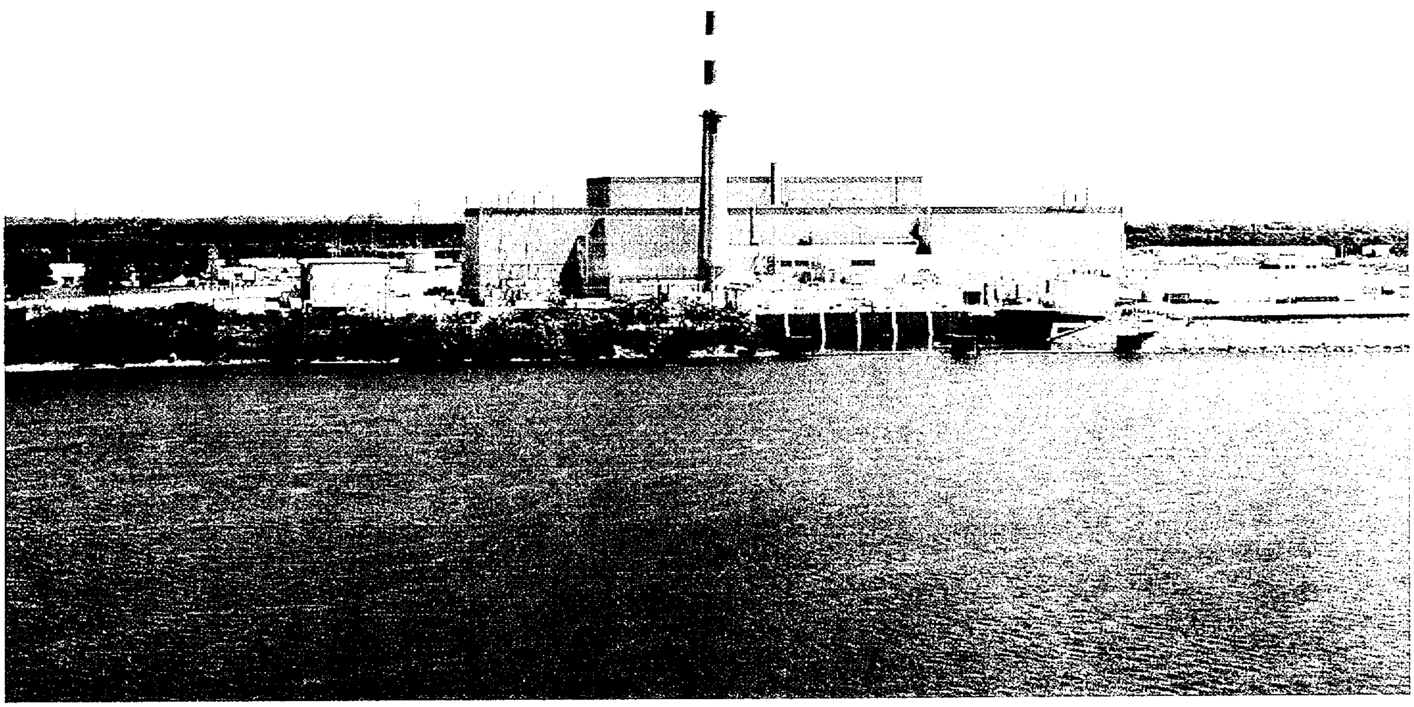
Improved Technical Specifications



Quad Cities Station

Volume 2:
Sections 3.1 and 3.2

ComEd

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 SDM shall be:

a. $\geq$ 0.38% $\Delta k/k$, with the highest worth control rod analytically determined; or

b. $\geq$ 0.28% $\Delta k/k$, with the highest worth control rod determined by test.

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 one standby gas treatment (SGT) subsystem 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 one SGT subsystem 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: a. $\geq$ 0.38% $\Delta k/k$ with the highest worth control rod analytically determined; or b. $\geq$ 0.28% $\Delta k/k$ with the highest worth control rod determined by test.	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 monitored core k_{eff} and the predicted core k_{eff} shall be within ± 1% $\Delta k/k$.

APPLICABILITY: MODES 1 and 2.

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 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	Verify core reactivity difference between the monitored core k_{eff} and the predicted core k_{eff} 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 operations in MODE 1

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Control Rod OPERABILITY

LCO 3.1.3 Each control rod shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each control rod.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One withdrawn control rod stuck.	------------NOTE------------ Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation," if required, to allow continued operation. ----------------------------	
	A.1 Verify stuck control rod separation criteria are met.	Immediately
	<u>AND</u>	
	A.2 Disarm the associated control rod drive (CRD).	2 hours
	<u>AND</u>	
		(continued)

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)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. ---------NOTE-------- Not applicable when THERMAL POWER > 10% RTP. --------------------- Two or more inoperable control rods not in compliance with analyzed rod position sequence and not separated by two or more OPERABLE control rods.	D.1 OR D.2	Restore compliance with analyzed rod position sequence. Restore control rod to OPERABLE status.	4 hours 4 hours
E. Required Action and associated Completion Time of Condition A, C, or D not met. OR Nine or more control rods inoperable.	E.1	Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Determine the position of each control rod.	24 hours
SR 3.1.3.2	------------------NOTE------------------ Not required to be performed until 7 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of RWM. -- Insert each fully withdrawn control rod at least one notch.	7 days
SR 3.1.3.3	------------------NOTE------------------ Not required to be performed until 31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM. -- Insert each partially withdrawn control rod at least one notch.	31 days
SR 3.1.3.4	Verify each control rod scram time from fully withdrawn to 90% insertion is $\leq$ 7 seconds.	In accordance with SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.3.5 Verify each control rod does not go to the withdrawn overtravel position.	Each time the control rod is withdrawn to "full out" position <u>AND</u> Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect coupling

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Control Rod Scram Times

LCO 3.1.4

a. No more than 12 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	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 each reactor shutdown $\geq$ 120 days

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
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
SR 3.1.4.3	Verify 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 fuel movement within the affected core cell <u>AND</u> 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 90% insertion. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

PERCENT INSERTION	SCRAM TIMES(a)(b) (seconds) when REACTOR STEAM DOME PRESSURE $\geq$ 800 psig
5	0.36
20	0.84
50	1.86
90	3.25

(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.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Control Rod Scram Accumulators

LCO 3.1.5 Each control rod scram accumulator shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each control rod scram accumulator.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One control rod scram accumulator inoperable with reactor steam dome pressure $\geq$ 900 psig.	A.1	--------NOTE-------- Only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time Surveillance. -------------------- Declare the associated control rod scram time "slow."	8 hours
	<u>OR</u>		
	A.2	Declare the associated control rod inoperable.	8 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two or more control rod scram accumulators inoperable with reactor steam dome pressure $\geq$ 900 psig.	B.1 Restore charging water header pressure to $\geq$ 940 psig.	20 minutes from discovery of Condition B concurrent with charging water header pressure < 940 psig
	AND	
	B.2.1 --------NOTE--------- Only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time Surveillance. -------------------- Declare the associated control rod scram time "slow."	1 hour
	OR	
	B.2.2 Declare the associated control rod inoperable.	1 hour

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more control rod scram accumulators inoperable with reactor steam dome pressure < 900 psig.	C.1	Verify all control rods associated with inoperable accumulators are fully inserted.	Immediately upon discovery of charging water header pressure < 940 psig
	AND		
	C.2	Declare the associated control rod inoperable.	1 hour
D. Required Action B.1 or C.1 and associated Completion Time not met.	D.1	--------NOTE--------- Not applicable if all inoperable control rod scram accumulators are associated with fully inserted control rods. -------------------- Place the reactor mode switch in the shutdown position.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify each control rod scram accumulator pressure is $\geq$ 940 psig.	7 days

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Rod Pattern Control

LCO 3.1.6 OPERABLE control rods shall comply with the requirements of the analyzed rod position sequence.

APPLICABILITY: MODES 1 and 2 with THERMAL POWER $\leq$ 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more OPERABLE control rods not in compliance with the analyzed rod position sequence.	A.1	--------NOTE-------- Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation." -------------------- Move associated control rod(s) to correct position.	8 hours
	<u>OR</u>		
	A.2	Declare associated control rod(s) inoperable.	8 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Nine or more OPERABLE control rods not in compliance with the analyzed rod position sequence.	B.1	--------NOTE--------- Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1. ---------------------	
		Suspend withdrawal of control rods.	Immediately
	AND		
	B.2	Place the reactor mode switch in the shutdown position.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.6.1	Verify all OPERABLE control rods comply with the analyzed rod position sequence.	24 hours

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. One SLC subsystem inoperable.	A.1 Restore SLC subsystem to OPERABLE status.	7 days
B. Two SLC subsystems inoperable.	B.1 Restore one SLC subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify available volume of sodium pentaborate solution is within the limits of Figure 3.1.7-1.	24 hours
SR 3.1.7.2	Verify temperature of sodium pentaborate solution is within the limits of Figure 3.1.7-2.	24 hours
SR 3.1.7.3	Verify temperature of pump suction piping is $\geq$ 83°F.	24 hours
SR 3.1.7.4	Verify continuity of explosive charge.	31 days
SR 3.1.7.5	Verify the concentration of sodium pentaborate in solution is within the limits of Figure 3.1.7-1.	31 days AND Once within 24 hours after water or sodium pentaborate is added to solution AND Once within 24 hours after solution temperature is restored within the limits of Figure 3.1.7-2

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.6	Verify each SLC subsystem manual 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	Verify each pump develops a flow rate $\geq$ 40 gpm at a discharge pressure $\geq$ 1275 psig.	In accordance with the Inservice Testing Program
SR 3.1.7.8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	24 months on a STAGGERED TEST BASIS
SR 3.1.7.9	Verify all heat traced piping between storage tank and pump suction is unblocked.	24 months <u>AND</u> Once within 24 hours after piping temperature is restored within the limits of Figure 3.1.7-2

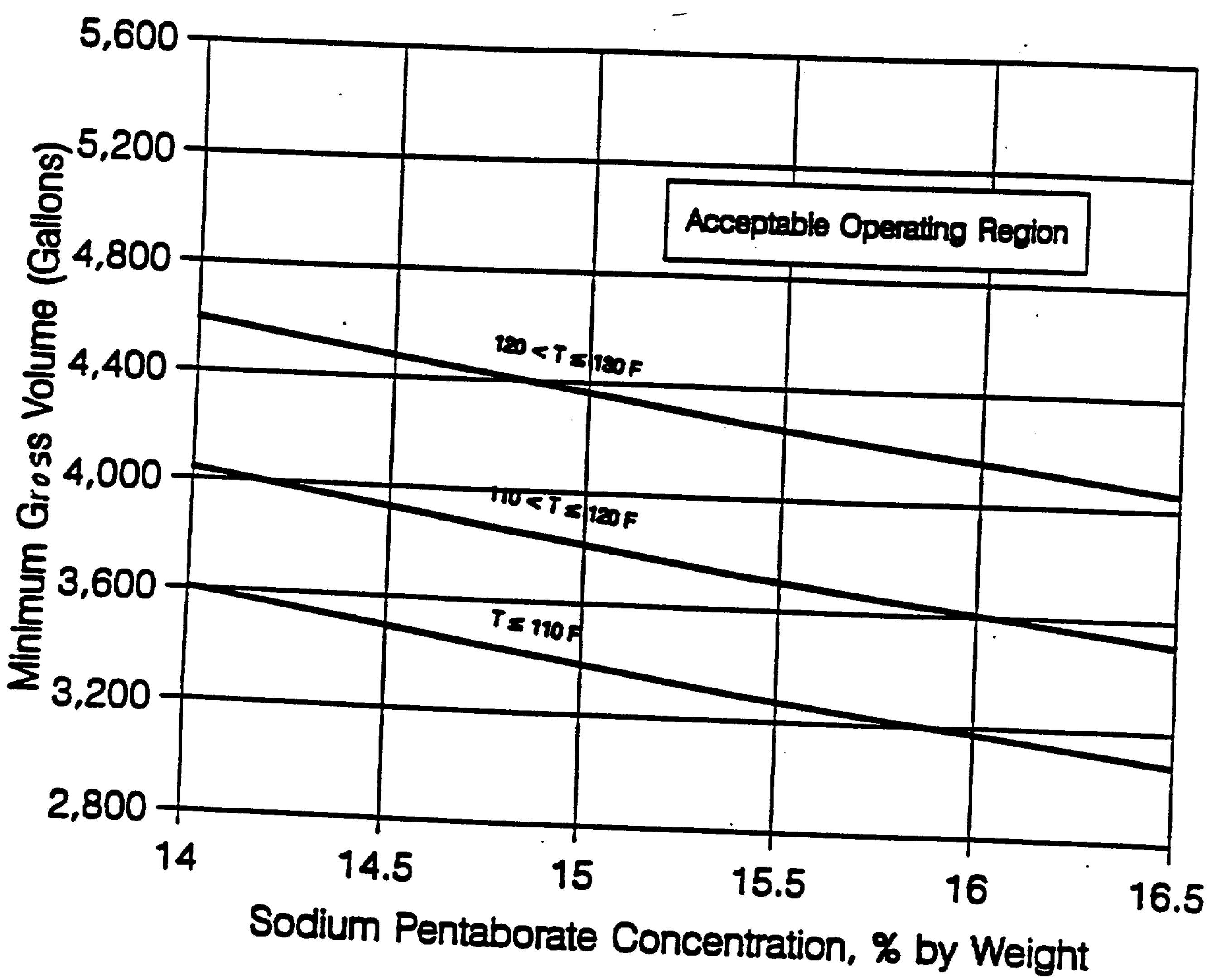


Figure 3.1.7-1 (page 1 of 1)
Sodium Pentaborate Volume Requirements

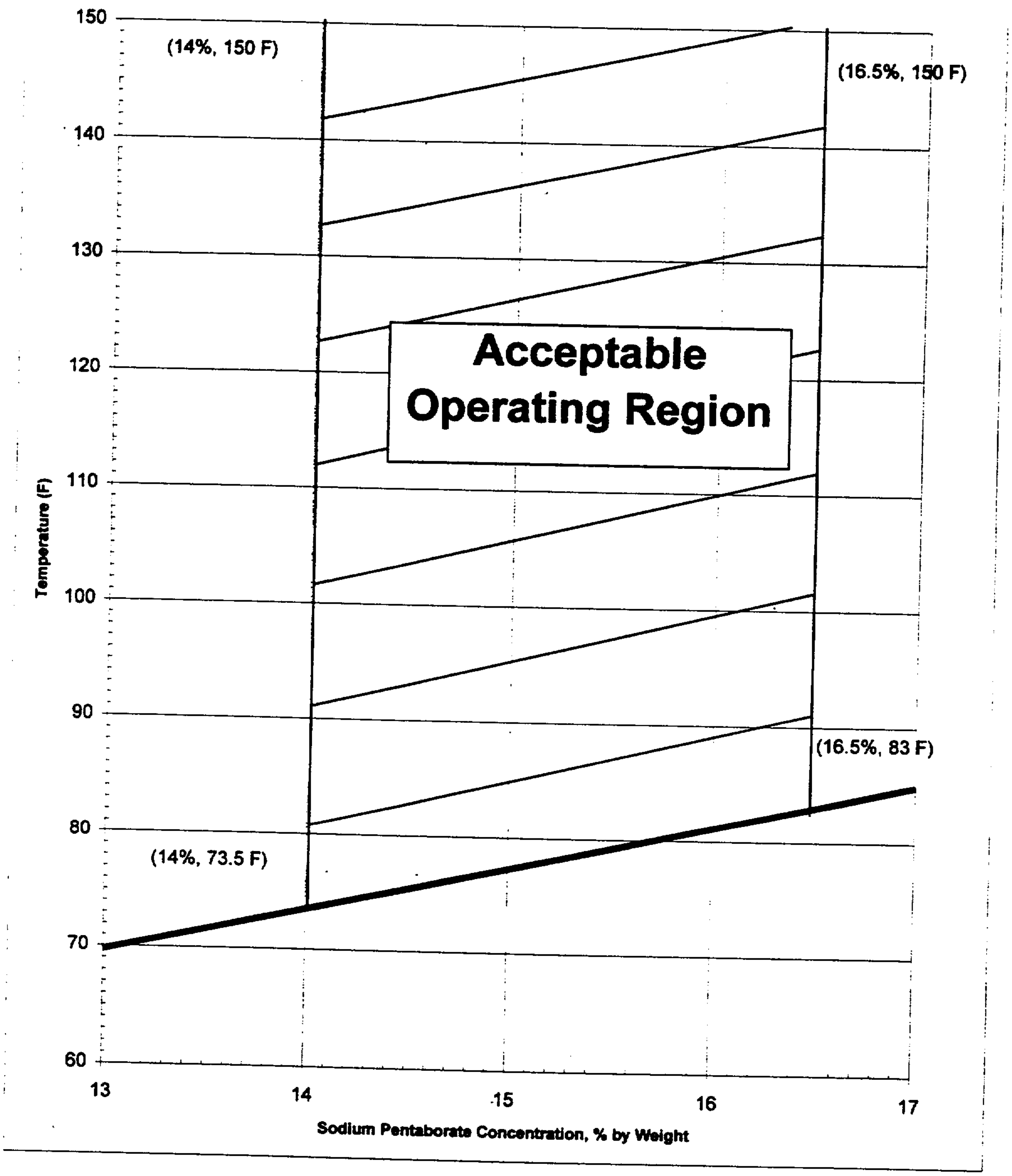


Figure 3.1.7-2 (page 1 of 1)
Sodium Pentaborate Temperature Requirements

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

LCO 3.1.8 Each SDV vent and drain valve shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

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

1. Separate Condition entry is allowed for each SDV vent and drain line.

2. An isolated line may be unisolated under administrative control to allow draining and venting of the SDV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SDV vent or drain lines with one valve inoperable.	A.1 Isolate the associated line.	7 days
B. One or more SDV vent or drain lines with both valves inoperable.	B.1 Isolate the associated line.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.8.1	------------------NOTE------------------ Not required to be met on vent and drain valves closed during performance of SR 3.1.8.2. -- Verify each SDV vent and drain valve is open.	31 days
SR 3.1.8.2	Cycle each SDV vent and drain valve to the fully closed and fully open position.	92 days
SR 3.1.8.3	Verify each SDV vent and drain valve: a. Closes in $\leq$ 30 seconds after receipt of an actual or simulated scram signal; and b. Opens when the actual or simulated scram signal is reset.	24 months

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

SDM requirements are specified to ensure:

a. The reactor can be made subcritical from all operating conditions and transients and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits; and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

These requirements are satisfied by the control rods, as described in UFSAR, Sections 3.1.5 and 4.6.2.1 (Ref. 1), which can compensate for the reactivity effects of the fuel and water temperature changes experienced during all operating conditions.

APPLICABLE SAFETY ANALYSES

Having sufficient SDM assures that the reactor will become and remain subcritical after all design basis accidents and transients. For example, SDM is assumed as an initial condition for the control rod removal error during refueling (Ref. 2) accident. The analysis of this reactivity insertion event assumes the refueling interlocks are OPERABLE when the reactor is in the refueling mode of operation. These interlocks prevent the withdrawal of more than one control rod from the core during refueling. (Special consideration and requirements for multiple control rod withdrawal during refueling are covered in Special Operations LCO 3.10.5, "Multiple Control Rod Withdrawal-Refueling.") The analysis assumes this condition is acceptable since the core will be shut down with the highest worth control rod withdrawn, if adequate SDM has been demonstrated.

Prevention or mitigation of positive reactivity insertion events is necessary to limit the energy deposition in the fuel, thereby preventing significant fuel damage, which

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	could result in undue release of radioactivity. Adequate SDM ensures inadvertent criticalities do not cause significant fuel damage. SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The specified SDM limit accounts for the uncertainty in the demonstration of SDM by testing. Separate SDM limits are provided for testing where the highest worth control rod is determined analytically or by measurement. This is due to the reduced uncertainty in the SDM test when the highest worth control rod is determined by measurement. When SDM is demonstrated by calculations not associated with a test (e.g., to confirm SDM during the fuel loading sequence), additional margin is included to account for uncertainties in the calculation. To ensure adequate SDM, a design margin is included to account for uncertainties in the design calculations (Ref. 3).
APPLICABILITY	In MODES 1 and 2, SDM must be provided to assure shutdown capability. 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 (Ref. 2).
ACTIONS	<u>A.1</u> 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.

(continued)

BASES

ACTIONS (continued)

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 Standby Gas Treatment (SGT) subsystem is OPERABLE; and secondary containment isolation capability is available in each associated secondary containment penetration flow path not isolated that is assumed to be isolated to mitigate radioactivity releases (i.e., at least one secondary containment isolation valve and associated instrumentation are OPERABLE, or other acceptable administrative controls to assure isolation capability). These administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated. This (ensuring components are OPERABLE) may be performed as an

(continued)

BASES

ACTIONS

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

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 one SGT subsystem is OPERABLE; and secondary containment isolation capability is available in each associated secondary containment penetration flow path not isolated that is assumed to be isolated to mitigate radioactivity releases (i.e., at least one secondary containment isolation valve and associated instrumentation are OPERABLE, or other acceptable administrative controls to

(continued)

BASES

ACTIONS

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

assure isolation capability). These administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated. This (ensuring components are OPERABLE) may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the Surveillances as needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, SRs may need to be performed to restore the component to OPERABLE status. Action must continue until all required components are OPERABLE.

SURVEILLANCE 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 by testing before or during the first startup after fuel movement, shuffling within the reactor pressure vessel, or control rod replacement. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control rod from another core location. Since core reactivity will vary during the cycle as a function of fuel depletion and poison burnup, the beginning of cycle (BOC) test must also account for changes in core reactivity during the cycle. Therefore, to obtain the SDM, the initial measured value 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 (Refs. 3 and 4). For

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

the SDM demonstrations that rely solely on calculation of the highest worth control rod, additional margin (0.10% $\Delta k/k$) must be added to the SDM limit of 0.28% $\Delta k/k$ to account for uncertainties in the calculation.

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

Local critical tests require the withdrawal of out of sequence control rods. This testing would therefore require bypassing of the rod worth minimizer to allow the out of sequence withdrawal, and therefore additional requirements must be met (see LCO 3.10.6, "Control Rod Testing – Operating").

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

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

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Sections 3.1.5 and 4.6.2.1.
2. UFSAR, Section 15.4.1.
3. UFSAR, Section 4.3.2.1.3.
4. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (as specified in Technical Specification 5.6.5).

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Anomalies

BASES

BACKGROUND

In accordance with UFSAR, Sections 3.1.5.1, 3.1.5.5, and 3.1.5.6 (Ref. 1), reactivity shall be controllable such that subcriticality is maintained under cold conditions and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, Reactivity Anomalies is used as a measure of the predicted versus measured core reactivity during power operation. The continual confirmation of core reactivity is necessary to ensure that the Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity anomaly could be the result of unanticipated changes in fuel reactivity or control rod worth or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in assuring the reactor can be brought safely to cold, subcritical conditions.

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

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

(continued)

BASES

BACKGROUND (continued)	absorbers (e.g., gadolinia), control rods, and whatever neutron poisons (mainly xenon and samarium) are present in the fuel. The predicted core reactivity, as represented by k effective (k_{eff}) is calculated by a 3D core simulator code as a function of cycle exposure. This calculation is performed for projected operating states and conditions throughout the cycle. The core reactivity is determined from k_{eff} for actual plant conditions and is then compared to the predicted value for the cycle exposure.
APPLICABLE SAFETY ANALYSES	Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations (Ref. 2). In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod drop accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity anomaly provides additional assurance that the nuclear methods provide an accurate representation of the core reactivity. The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted core K_{eff} for identical core conditions at BOC do not reasonably agree, then the assumptions used in the reload cycle design analysis or the calculation models used to predict core k_{eff} may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured value. Thereafter, any significant deviations in the measured core k_{eff} from the predicted core k_{eff} that develop during fuel depletion may be an indication that the assumptions of the DBA and transient analyses are no longer valid, or that an unexpected change in core conditions has occurred. Reactivity Anomalies satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

(continued)

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 core k_{eff} of $\pm$ 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. 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, Reactivity Anomalies is not required during these conditions.
ACTIONS	<u>A.1</u> Should an anomaly develop between measured and 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 conditions to determine their consistency with input to design calculations. Measured core and process parameters

(continued)

BASES

ACTIONS

A.1 (continued)

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 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 3.1.2.1

Verifying the reactivity difference between the monitored and predicted core k_{eff} 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 core k_{eff} for the reactor conditions obtained from plant instrumentation. A comparison of the monitored core k_{eff} to the predicted core k_{eff} 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 rod from another core location. Also, core reactivity changes during the cycle. The 24 hour interval after reaching equilibrium

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1 (continued)

conditions following a startup is based on the need for equilibrium xenon concentrations in the core, such that an accurate comparison between the monitored and predicted core k_{eff} can be made. For the purposes of this SR, the reactor is assumed to be at equilibrium conditions when steady state operations (no control rod movement or core flow changes) at $\geq$ 75% RTP have been obtained. The 1000 MWD/T Frequency was developed, considering the relatively slow change in core reactivity with exposure and operating experience related to variations in core reactivity. This comparison requires the core to be operating at power levels which minimize the uncertainties and measurement errors, in order to obtain meaningful results. Therefore, the comparison is only done when in MODE 1. The core weight, tons(T) in MWD/T, reflects metric tons.

REFERENCES

1. UFSAR, Sections 3.1.5.1, 3.1.5.5, and 3.1.5.6.

2. UFSAR, Chapter 15.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Control Rod OPERABILITY

BASES

BACKGROUND

Control rods are components of the control rod drive (CRD) System, which is the primary reactivity control system for the reactor. In conjunction with the Reactor Protection System, the CRD System provides the means for the reliable control of reactivity changes to ensure under conditions of normal operation, including anticipated operational occurrences, that specified acceptable fuel design limits are not exceeded. In addition, the control rods provide the capability to hold the reactor core subcritical under all conditions and to limit the potential amount and rate of reactivity increase caused by a malfunction in the CRD System. The CRD System is designed to satisfy the requirements of UFSAR, Sections 3.1.5.1, 3.1.5.2, 3.1.5.3, 3.1.5.4, 3.1.5.5, and 3.1.5.6 (Ref. 1).

The CRD System consists of 177 locking piston control rod drive mechanisms (CRDMs) and a hydraulic control unit for each drive mechanism. The locking piston type CRDM is a double acting hydraulic piston, which uses contaminated condensate storage tank, fuel pool reject, or condensate water as the operating fluid. Accumulators provide additional energy for scram. An index tube and piston, coupled to the control rod, are locked at fixed increments by a collet mechanism. The collet fingers engage notches in the index tube to prevent unintentional withdrawal of the control rod, but without restricting insertion.

This Specification, along with LCO 3.1.4, "Control Rod Scram Times," LCO 3.1.5, "Control Rod Scram Accumulators," and LCO 3.1.6, "Rod Pattern Control," ensure that the performance of the control rods in the event of a Design Basis Accident (DBA) or transient meets the assumptions used in the safety analyses of References 2, 3, and 4.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in the evaluations involving control rods are presented in Reference 5. The control rods provide the primary means for rapid reactivity control (reactor scram), for maintaining

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

the reactor subcritical and for limiting the potential effects of reactivity insertion events caused by malfunctions in the CRD System.

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

The control rods also protect the fuel from damage which could result in release of radioactivity. The limits protected are the MCPR Safety Limit (SL) (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," and LCO 3.2.4, Average Power Range Monitor (APRM) Gain and Setpoint"), and the fuel design limit (see Bases for LCO 3.1.6, "Rod Pattern Control") during reactivity insertion events.

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

Control rod OPERABILITY satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The OPERABILITY of an individual control rod is based on a combination of factors, primarily, the scram insertion times, the control rod coupling integrity, and the ability

(continued)

BASES

LCO (continued)

to determine the control rod position. Accumulator OPERABILITY is addressed by LCO 3.1.5. The associated scram accumulator status for a control rod only affects the scram insertion times; therefore, an inoperable accumulator does not immediately require declaring a control rod inoperable. Although not all control rods are required to be OPERABLE to satisfy the intended reactivity control requirements, strict control over the number and distribution of inoperable control rods is required to satisfy the assumptions of the DBA and transient analyses.

OPERABILITY requirements for control rods also include correct assembly of the CRD housing supports.

APPLICABILITY

In MODES 1 and 2, the control rods are assumed to function during a DBA or transient and are therefore required to be OPERABLE in these MODES. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod OPERABILITY during these conditions. Control rod requirements in MODE 5 are located in LCO 3.9.5, "Control Rod OPERABILITY – Refueling."

ACTIONS

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

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

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

(continued)

BASES

ACTIONS

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

RWM is bypassed to ensure compliance with the CRDA analysis. With one withdrawn control rod stuck, the local scram reactivity rate assumptions may not be met if the stuck control rod separation criteria are not met. Therefore, a verification that the separation criteria are met must be performed immediately. The stuck control rod separation criteria are not met if: a) the stuck control rod occupies a location adjacent to two "slow" control rods, b) the stuck control rod occupies a location adjacent to one "slow" control rod, and the one "slow" control rod is also adjacent to another "slow" control rod, or c) if the stuck control rod occupies a location adjacent to one "slow" control rod when there is another pair of "slow" control rods elsewhere in the core adjacent to one another. The description of "slow" control rods is provided in LCO 3.1.4 "Control Rod Scram Times." In addition, the associated control rod drive must be disarmed in 2 hours. The allowed Completion Time of 2 hours is acceptable, considering the reactor can still be shut down, assuming no additional control rods fail to insert, and provides a reasonable time to perform the Required Action in an orderly manner. The control rod must be isolated from both scram and normal insert and withdraw pressure. Isolating the control rod from scram and normal insert and withdraw pressure prevents damage to the CRDM or reactor internals. The control rod isolation method should also ensure cooling water to the CRD is maintained.

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

(continued)

BASES

ACTIONS

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

the RWM (LCO 3.3.2.1). The allowed Completion Time provides a reasonable time to test the control rods, considering the potential for a need to reduce power to perform the tests.

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 MODE 3 conditions.

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.

(continued)

BASES

ACTIONS (continued)

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 to prevent inadvertent withdrawal during subsequent operations. The control rods can be hydraulically disarmed by closing the drive water and exhaust water isolation valves. 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.

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.

D.1 and D.2

Out of sequence control rods may increase the potential reactivity worth of a dropped control rod during a CRDA. At ≤ 10% RTP, the analyzed rod position sequence analysis (Refs. 6 and 7) requires inserted control rods not in compliance with the analyzed rod position sequence to be separated by at least two OPERABLE control rods in all directions, including the diagonal (i.e., all other control rods in a five-by-five array centered on the inoperable control rod are OPERABLE). Therefore, if two or more inoperable control rods are not in compliance with the analyzed rod position sequence and not separated by at least two OPERABLE control rods in all directions, action must be taken to restore compliance with the analyzed rod position sequence or restore the control rods to OPERABLE status. Condition D is modified by a Note indicating that the Condition is not applicable when > 10% RTP, since the analyzed rod position sequence is not required to be

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

followed under these conditions, as described in the Bases for LCO 3.1.6. The allowed Completion Time of 4 hours is acceptable, considering the low probability of a CRDA occurring.

E.1

If any Required Action and associated Completion Time of Condition A, C, or D are not met, or there are nine or more inoperable control rods, 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. This ensures all insertable control rods are inserted and places the reactor in a condition that does not require the active function (i.e., scram) of the control rods. The number of control rods permitted to be inoperable when operating above 10% RTP (e.g., no CRDA considerations) could be more than the value specified, but the occurrence of a large number of inoperable control rods could be indicative of a generic problem, and investigation and resolution of the potential problem should be undertaken. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

The position of each control rod must be determined to ensure adequate information on control rod position is available to the operator for determining control rod OPERABILITY and controlling rod patterns. Control rod position may be determined by the use of OPERABLE position indicators, by moving control rods to a position with an OPERABLE indicator (full-in, full-out, or numeric indicators), by verifying the indicators one notch "out" and one notch "in" are OPERABLE, or by the use of other appropriate methods. The 24 hour Frequency of this SR is based on operating experience related to expected changes in control rod position and the availability of control rod position indications in the control room.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.3.2 and SR 3.1.3.3

Control rod insertion capability is demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. The control rod may then be returned to its original position. This ensures the control rod is not stuck and is free to insert on a scram signal. These Surveillances are not required when THERMAL POWER is less than or equal to the actual LPSP of the RWM, since the notch insertions may not be compatible with the requirements of the analyzed rod position sequence (LCO 3.1.6) and the RWM (LCO 3.3.2.1). The 7 day Frequency of SR 3.1.3.2 is based on operating experience related to the changes in CRD performance and the ease of performing notch testing for fully withdrawn control rods. Partially withdrawn control rods are tested at a 31 day Frequency, based on the potential power reduction required to allow the control rod movement and considering the large testing sample of SR 3.1.3.2. Furthermore, the 31 day Frequency takes into account operating experience related to changes in CRD performance. At any time, if a control rod is immovable, a determination of that control rod's trippability (OPERABILITY) must be made and appropriate action taken.

These SRs are modified by Notes that allow 7 days and 31 days respectively, after withdrawal of the control rod and increasing power to above the LPSP, to perform the Surveillance. This acknowledges that the control rod must be first withdrawn and THERMAL POWER must be increased to above the LPSP before performance of the Surveillance, and therefore, the Notes avoid potential conflicts with SR 3.0.3 and SR 3.0.4.

SR 3.1.3.4

Verifying that the scram time for each control rod to 90% insertion is $\leq$ 7 seconds provides reasonable assurance that the control rod will insert when required during a DBA or transient, thereby completing its shutdown function. This SR is performed in conjunction with the control rod scram time testing of SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1, "Reactor Protection System (RPS)

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.3.4 (continued)

Instrumentation," and the functional testing of SDV vent and drain valves in LCO 3.1.8, "Scram Discharge Volume (SDV) Vent and Drain Valves," overlap this Surveillance to provide complete testing of the assumed safety function. The associated Frequencies are acceptable, considering the more frequent testing performed to demonstrate other aspects of control rod OPERABILITY and operating experience, which shows scram times do not significantly change over an operating cycle.

SR 3.1.3.5

Coupling verification is performed to ensure the control rod is connected to the CRDM and will perform its intended function when necessary. The Surveillance requires verifying that a control rod does not go to the withdrawn overtravel position when it is fully withdrawn. The overtravel position feature provides a positive check on the coupling integrity since only an uncoupled CRD can reach the overtravel position. The verification is required to be performed any time a control rod is withdrawn to the "full out" position (notch position 48) or prior to declaring the control rod OPERABLE after work on the control rod or CRD System that could affect coupling. This includes control rods inserted one notch and then returned to the "full out" position during the performance of SR 3.1.3.2. This Frequency is acceptable, considering the low probability that a control rod will become uncoupled when it is not being moved and operating experience related to uncoupling events.

REFERENCES

1. UFSAR, Sections 3.1.5.1, 3.1.5.2, 3.1.5.3, 3.1.5.4, 3.1.5.5, and 3.1.5.6.

2. UFSAR, Section 5.2.2.2.3.

3. UFSAR, Section 6.2.1.3.2.

4. UFSAR, Chapter 15.

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BASES

REFERENCES (continued)

5. UFSAR, Section 4.6.3.4.2.1.
6. NEDO-21231, "Banked Position Withdrawal Sequence," Section 7.2, January 1977.
7. NFSR-0091, Commonwealth Edison Topical Report, Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods, (as specified in Technical Specification 5.6.5).

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Control Rod Scram Times

BASES

BACKGROUND	The scram function of the Control Rod Drive (CRD) System controls reactivity changes during anticipated operational occurrences to ensure that specified acceptable fuel design limits are not exceeded (Ref. 1). The control rods are scrammed by positive means using hydraulic pressure exerted on the CRD piston. When a scram signal is initiated, control air is vented from the scram valves, allowing them to open by spring action. Opening the exhaust valve reduces the pressure above the main drive piston to atmospheric pressure, and opening the inlet valve applies the accumulator or reactor pressure to the bottom of the piston. Since the notches in the index tube are tapered on the lower edge, the collet fingers are forced open by cam action, allowing the index tube to move upward without restriction because of the high differential pressure across the piston. As the drive moves upward and the accumulator pressure reduces below the reactor pressure, a ball check valve opens, letting the reactor pressure complete the scram action. If the reactor pressure is low, such as during startup, the accumulator will fully insert the control rod in the required time without assistance from reactor pressure.
APPLICABLE SAFETY ANALYSES	The analytical methods and assumptions used in evaluating the control rod scram function are presented in Reference 2. The Design Basis Accident (DBA) and transient analyses assume that all of the control rods scram at a specified insertion rate. The resulting negative scram reactivity forms the basis for the determination of plant thermal limits (e.g., the MCPR). Other distributions of scram times (e.g., several control rods scramming slower than the average time with several control rods scramming faster than the average time) can also provide sufficient scram reactivity. Surveillance of each individual control rod's scram time ensures the scram reactivity assumed in the DBA and transient analyses can be met.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

The scram function of the CRD System protects the MCPR Safety Limit (SL) (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," and LCO 3.2.4, Average Power Range Monitor (APRM) Gain and Setpoint"), which ensure that no fuel damage will occur if these limits are not exceeded. At ≥ 800 psig, the scram function is designed to insert negative reactivity at a rate fast enough to prevent the actual MCPR from becoming less than the MCPR SL, during the analyzed limiting power transient. Below 800 psig, the scram function is assumed to perform during the control rod drop accident (Ref. 3) and, therefore, also provides protection against violating fuel design limits during reactivity insertion accidents (see Bases for LCO 3.1.6, "Rod Pattern Control"). For the reactor vessel overpressure protection analysis, the scram function, along with the safety/relief valves, ensure that the peak vessel pressure is maintained within the applicable ASME Code limits.

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

LCO

The scram times specified in Table 3.1.4-1 are required to ensure that the scram reactivity assumed in the DBA and transient analysis is met (Ref. 4). To account for single failures and "slow" scramming control rods, the scram times specified in Table 3.1.4-1 are faster than those assumed in the design basis analysis. The scram times have a margin that allows up to approximately 7% of the control rods (e.g., 177 x 7% ≈ 12) to have scram times exceeding the specified limits (i.e., "slow" control rods) assuming a single stuck control rod (as allowed by LCO 3.1.3, "Control Rod OPERABILITY") and an additional control rod failing to scram per the single failure criterion. The scram times are specified as a function of reactor steam dome pressure to account for the pressure dependence of the scram times. The scram times are specified relative to measurements based on reed switch positions, which provide the control rod position indication. The reed switch closes ("pickup") when the index tube passes a specific location and then opens

(continued)

BASES

LCO (continued)

("dropout") as the index tube travels upward. Verification of the specified scram times in Table 3.1.4-1 is accomplished through measurement and interpolation of the "pickup" or "dropout" times of reed switches associated with each of the required insertion positions. To ensure that local scram reactivity rates are maintained within acceptable limits, no more than two of the allowed "slow" control rods may occupy adjacent locations (face or diagonal).

Table 3.1.4-1 is modified by two Notes which state that control rods with scram times not within the limits of the table are considered "slow" and that control rods with scram times > 7 seconds are considered inoperable as required by SR 3.1.3.4.

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

APPLICABILITY

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

ACTIONS

A.1

When the requirements of this LCO are not met, the rate of negative reactivity insertion during a scram may not be within the assumptions of the safety analyses. Therefore, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating

(continued)

BASES

ACTIONS

A.1 (continued)

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 5, 6, and 7.

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

(continued)

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 (i.e., 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

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 the affected control rod is still within acceptable limits. The scram time limits for reactor pressures < 800 psig are found in the Technical Requirements Manual (Ref. 8) and 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."

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.3 (continued)

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

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

SR 3.1.4.4

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

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

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 3.1.
2. UFSAR, Section 4.6.3.4.2.1.
3. UFSAR, Section 15.4.10.
4. 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.
5. UFSAR, Section 5.2.2.2.3.
6. UFSAR, Section 6.2.1.3.2.
7. UFSAR, Chapter 15.
8. Technical Requirements Manual.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Control Rod Scram Accumulators

BASES

BACKGROUND

The control rod scram accumulators are part of the Control Rod Drive (CRD) System and are provided to ensure that the control rods scram under varying reactor conditions. The control rod scram accumulators store sufficient energy to fully insert a control rod at any reactor vessel pressure. The accumulator is a hydraulic cylinder with a free floating piston. The piston separates the water used to scram the control rods from the nitrogen, which provides the required energy. The scram accumulators are necessary to scram the control rods within the required insertion times of LCO 3.1.4, "Control Rod Scram Times."

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the control rod scram function are presented in Reference 1. The Design Basis Accident (DBA) and transient analyses assume that all of the control rods scram at a specified insertion rate. OPERABILITY of each individual control rod scram accumulator, along with LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.4, ensures that the scram reactivity assumed in the DBA and transient analyses can be met. The existence of an inoperable accumulator may invalidate prior scram time measurements for the associated control rod.

The scram function of the CRD System, and therefore the OPERABILITY of the accumulators, protects the MCPR Safety Limit (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," and LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoint"), which ensure that no fuel damage will occur if these limits are not exceeded (see Bases for LCO 3.1.4). In addition, the scram function at low reactor vessel pressure (i.e., startup conditions) provides protection against violating fuel design limits during reactivity insertion accidents (see Bases for LCO 3.1.6, "Rod Pattern Control").

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	Control rod scram accumulators satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The OPERABILITY of the control rod scram accumulators is required to ensure that adequate scram insertion capability exists when needed over the entire range of reactor pressures. The OPERABILITY of the scram accumulators is based on maintaining adequate accumulator pressure.
APPLICABILITY	In MODES 1 and 2, the scram function is required for mitigation of DBAs and transients, and therefore the scram accumulators must be OPERABLE to support the scram function. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod scram accumulator OPERABILITY during these conditions. Requirements for scram accumulators in MODE 5 are contained in LCO 3.9.5, "Control Rod OPERABILITY - Refueling."
ACTIONS	The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each control rod 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. <u>A.1 and A.2</u> With one control rod scram accumulator inoperable and the reactor steam dome pressure ≥ 900 psig, 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 in Table 3.1.4-1. Required Action A.1 is modified by a Note indicating that declaring the control rod "slow" only applies if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

scram time Surveillance. Otherwise, the control rod may already be considered "slow" and the further degradation of scram performance with an inoperable accumulator could result in excessive scram times. In this event, the associated control rod is declared inoperable (Required Action A.2) and LCO 3.1.3 is entered. This would result in requiring the affected control rod to be fully inserted and disarmed, thereby satisfying its intended function, in accordance with ACTIONS of LCO 3.1.3.

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

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

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

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

(continued)

BASES

ACTIONS

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

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

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

C.1 and C.2

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

D.1

The reactor mode switch must be immediately placed in the shutdown position if either Required Action and associated Completion Time associated with loss of the CRD pump (Required Actions B.1 and C.1) cannot be met. This ensures

(continued)

BASES

ACTIONS

D.1 (continued)

that all insertable control rods are inserted and that the reactor is in a condition that does not require the active function (i.e., scram) of the control rods. This Required Action is modified by a Note stating that the action is not applicable if all control rods associated with the inoperable scram accumulators are fully inserted, since the function of the control rods has been performed.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

SR 3.1.5.1 requires that the accumulator pressure be checked every 7 days to ensure adequate accumulator pressure exists to provide sufficient scram force. The primary indicator of accumulator OPERABILITY is the accumulator pressure. A minimum accumulator pressure is specified, below which the capability of the accumulator to perform its intended function becomes degraded and the accumulator is considered inoperable. The minimum accumulator pressure of 940 psig is well below the expected pressure of 1100 psig (Ref. 2). Declaring the accumulator inoperable when the minimum pressure is not maintained ensures that significant degradation in scram times does not occur. The 7 day Frequency has been shown to be acceptable through operating experience and takes into account indications available in the control room.

REFERENCES

1. UFSAR, Section 4.6.3.4.2.1.
2. Letter, from E.Y. Gibo (GE) to P Chenell (ComEd), "Generic Basis for HCU Scram Accumulator Minimum Setpoint Pressure," April 10, 1998.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Rod Pattern Control

BASES

BACKGROUND

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

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

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the CRDA are summarized in References 1, 2, 3, 4, and 5. CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analysis. The RWM (LCO 3.3.2.1) provides backup to operator control of the withdrawal sequences to ensure that the initial conditions of the CRDA analysis are not violated.

Prevention or mitigation of positive reactivity insertion events is necessary to limit the energy deposition in the fuel, thereby preventing significant fuel damage which could result in the undue release of radioactivity. Since the failure consequences for UO_2 have been shown to be insignificant below fuel energy depositions of 300 cal/gm (Ref. 6), the fuel design limit of 280 cal/gm provides a margin of safety from significant core damage which would result in release of radioactivity (Ref. 7). Generic evaluations (Refs. 8 and 9) of a design basis CRDA (i.e., a CRDA resulting in a peak fuel energy deposition of 280 cal/gm) have shown that if the peak fuel enthalpy remains below 280 cal/gm, then the maximum reactor pressure will be less than the required ASME Code limits (Ref. 10) and the calculated offsite doses will be well within the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

required limits (Ref. 11). Cycle specific CRDA analyses are performed that assume eight inoperable control rods with at least two cell separation and confirm fuel energy deposition is less than 280 cal/gm.

Control rod patterns analyzed in the cycle specific analyses follow predetermined sequencing rules (analyzed rod position sequence). The analyzed rod position sequence is applicable from the condition of all control rods fully inserted to 10% RTP (Ref. 5). The control rods are required to be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions (e.g., between notches 08 and 12). The banked positions are established to minimize the maximum incremental control rod worth without being overly restrictive during normal plant operation. Cycle specific analyses ensure that the 280 cal/gm fuel design limit will not be violated during a CRDA under worst case scenarios. The cycle specific analyses (Refs. 1, 2, 3, 4, and 5) also evaluate the effect of fully inserted, inoperable control rods not in compliance with the sequence, to allow a limited number (i.e., eight) and distribution of fully inserted, inoperable control rods. Specific analysis may also be performed for atypical operating conditions (e.g., fuel leaker suppression).

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

LCO

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

APPLICABILITY

In MODES 1 and 2, when THERMAL POWER is ≤ 10% RTP, the CRDA is a Design Basis Accident and, therefore, compliance with the assumptions of the safety analysis is required. When THERMAL POWER is > 10% RTP, there is no credible control rod configuration that results in a control rod worth that could

(continued)

BASES

APPLICABILITY (continued)

exceed the 280 cal/gm fuel design limit during a CRDA (Refs. 4 and 5). In MODES 3 and 4, the reactor is shutdown and the control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied, therefore, a CRDA is not postulated to occur. In MODE 5, since the reactor is shut down and only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will remain subcritical with a single control rod withdrawn.

ACTIONS

A.1 and A.2

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

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

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

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

When nine or more OPERABLE control rods are not in compliance with the analyzed rod position sequence, 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 analyzed rod position sequence 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 analyzed rod position sequence is performed by the RWM (LCO 3.3.2.1), which provides control rod blocks to enforce the required sequence and is required to be OPERABLE when operating at $\leq$ 10% RTP.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 15.4.10.

2. XN-NF-80-19(P)(A), Volume 1, Supplement 2, Section 7.1 Exxon Nuclear Methodology for Boiling Water Reactor-Neutronics Methods for Design and Analysis, (as specified in Technical Specification 5.6.5).

3. NEDE-24011-P-A, "GE Standard Application for Reactor Fuel," (as specified in Technical Specification 5.6.5).

4. Letter from T.A. Pickens (BWROG) to G.C. Lainas (NRC), "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A," BWROG-8644, August 15, 1986.

5. NFSR-0091, Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods, Commonwealth Edison Topical Report, (as specified in Technical Specification 5.6.5).

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

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

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

9. NEDO-10527, "Rod Drop Accident Analysis for Large BWRs," (including Supplements 1 and 2), March 1972.

10. ASME, Boiler and Pressure Vessel Code.

11. 10 CFR 100.11.

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

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

BASES

BACKGROUND	The SLC System is designed to provide the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory (which is at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive, xenon free state without taking credit for control rod movement. The SLC System satisfies the requirements of 10 CFR 50.62 (Ref. 1) on anticipated transient without scram. The SLC System consists of a boron solution storage tank, two positive displacement pumps, two explosive valves that are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged near the bottom of the core shroud, where it then mixes with the cooling water rising through the core. A smaller tank containing demineralized water is provided for testing purposes.
APPLICABLE SAFETY ANALYSES	The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator determines the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System is used in the event that enough control rods cannot be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to add negative reactivity to compensate for all of the various reactivity effects that could occur during plant operations. To meet this objective, it is necessary to inject a quantity of boron, which produces a concentration of 600 ppm of natural boron, in the reactor coolant at 68°F. To allow for potential leakage and imperfect mixing in the reactor system, an amount of boron equal to 25% of the amount cited above is added (Ref. 2). The volume versus concentration limits in Figure 3.1.7-1 and the temperature versus concentration limits in Figure 3.1.7-2 are calculated such that the required concentration is achieved accounting for dilution in the RPV with reactor water level at the high alarm point, including the water volume in the residual heat removal shutdown

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

cooling piping, the recirculation loop piping, and portions of other piping systems which connect to the RPV below the high alarm point. This quantity of borated solution represented is the amount that is above the bottom of the boron solution storage tank. However, no credit is taken for the portion of the tank volume that cannot be injected.

The SLC System satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The OPERABILITY of the SLC System provides backup capability for reactivity control independent of normal reactivity control provisions provided by the control rods. The OPERABILITY of the SLC System is based on the conditions of the borated solution in the storage tank and the availability of a flow path to the RPV, including the OPERABILITY of the pumps and valves. Two SLC subsystems are required to be OPERABLE; each contains an OPERABLE pump, an explosive valve, and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path. With one subsystem inoperable the requirements of 10 CFR 50.62 (Ref. 1) cannot be met, however, the remaining subsystem is still capable of shutting down the unit.

APPLICABILITY

In MODES 1 and 2, shutdown capability is required. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate controls to ensure that the reactor remains subcritical. In MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Demonstration of adequate SDM (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") ensures that the reactor will not become critical. Therefore, the SLC System is not required to be OPERABLE when only a single control rod can be withdrawn.

ACTIONS

A.1

If one SLC subsystem is inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE subsystem is adequate to

(continued)

BASES

ACTIONS

A.1 (continued)

shutdown the unit. However, the overall capability is reduced since the remaining OPERABLE subsystem cannot meet the requirements of Reference 1. The 7 day Completion Time is based on the availability of an OPERABLE subsystem capable of shutting down the reactor and the low probability of a Design Basis Accident (DBA) or severe transient occurring concurrent with the failure of the Control Rod Drive (CRD) System to shut down the reactor.

B.1

If both SLC subsystems are inoperable, at least one subsystem must be restored to OPERABLE status within 8 hours. The allowed Completion Time of 8 hours is considered acceptable given the low probability of a DBA or transient occurring concurrent with the failure of the control rods to shut down the reactor.

C.1

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to 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

SR 3.1.7.1 through SR 3.1.7.3 are 24 hour Surveillances 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 Surveillances ensure that the proper borated solution volume and temperature, including the temperature of the pump suction piping, are maintained. Maintaining a minimum specified borated solution temperature is important in ensuring that the boron remains in solution and does not

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1, SR 3.1.7.2, and SR 3.1.7.3 (continued)

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 has shown there are relatively slow variations in the measured parameters of volume and temperature.

SR 3.1.7.4 and SR 3.1.7.6

SR 3.1.7.4 verifies the continuity of the explosive charges in the injection valves to ensure that proper operation will occur if required. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day Frequency is based on operating experience and has demonstrated the reliability of the explosive charge continuity.

SR 3.1.7.6 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 manual 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.5

This Surveillance requires an examination of the sodium pentaborate solution by using chemical analysis to ensure that the proper concentration of sodium pentaborate exists in the storage tank. SR 3.1.7.5 must be performed anytime boron or water is added to the storage tank solution to determine that the sodium pentaborate 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 sodium pentaborate concentration between surveillances.

SR 3.1.7.7

Demonstrating that each SLC System pump develops a flow rate ≥ 40 gpm at a discharge pressure ≥ 1275 psig ensures that pump performance has not degraded during the fuel cycle. This minimum pump flow rate requirement ensures that, when combined with the sodium pentaborate solution concentration requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, and detect incipient failures by indicating abnormal performance. The Frequency of this Surveillance is in accordance with the Inservice Testing Program.

SR 3.1.7.8 and SR 3.1.7.9

These Surveillances ensure that there is a functioning flow path from the boron solution storage tank to the RPV, including the firing of an explosive valve. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. The pump and explosive valve tested

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.8 and SR 3.1.7.9 (continued)

should be alternated such that both complete flow paths are tested every 48 months at alternating 24 month intervals. The Surveillance may be performed in separate steps to prevent injecting boron into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a test tank through one SLC subsystem and into the RPV. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

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

The 24 month Frequency is acceptable since there is a low probability that the subject piping will be blocked due to precipitation of the boron from solution in the heat traced piping. This is especially true in light of the temperature verification of this piping required by SR 3.1.7.3. However, if, in performing SR 3.1.7.3, it is determined that the temperature of this piping has fallen below the specified minimum, SR 3.1.7.9 must be performed once within 24 hours after the piping temperature is restored to within the limits of Figure 3.1.7-2.

REFERENCES

1. 10 CFR 50.62.
2. UFSAR, Section 9.3.5.3.

B 3.1 REACTIVITY CONTROL SYSTEMS

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

BASES

BACKGROUND

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

APPLICABLE SAFETY ANALYSES

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

a. Close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within the limits of 10 CFR 100 (Ref. 2); and

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

Isolation of the SDV can also be accomplished by manual closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. For a bounding leakage case, the offsite doses are well within the limits of 10 CFR 100 (Ref. 2), and adequate core cooling is maintained (Ref. 3). The SDV vent and drain valves allow continuous drainage of the SDV during normal plant operation

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

to ensure that the SDV has sufficient capacity to contain the reactor coolant discharge during a full core scram. To automatically ensure this capacity, a reactor scram (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation") is initiated if the SDV water level in the instrument volume exceeds a specified setpoint. The setpoint is chosen so that all control rods are inserted before the SDV has insufficient volume to accept a full scram.

SDV vent and drain valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

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

APPLICABILITY

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

ACTIONS

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

(continued)

BASES

ACTIONS (continued)

The ACTIONS Table is modified by a second Note stating that an isolated line may be unisolated under administrative control to allow draining and venting of the SDV. When a line is isolated, the potential for an inadvertent scram due to high SDV level is increased. During these periods, the line may be unisolated under administrative control. This allows any accumulated water in the line to be drained, to preclude a reactor scram on SDV high level. This is acceptable since the administrative controls ensure the valve can be closed quickly, by a dedicated operator at the valve controls, if a scram occurs with the valve open.

A.1

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

B.1

If both valves in a line are inoperable, the line must be isolated to contain the reactor coolant during a scram. The 8 hour Completion Time to isolate the line is based on the low probability of a scram occurring while the line is not isolated and unlikelihood of significant CRD seal leakage.

C.1

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. The allowed Completion

(continued)

BASES

ACTIONS

C.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

SR 3.1.8.1

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

The 31 day Frequency is based on engineering judgment and is consistent with the procedural controls governing valve operation, which ensure correct valve positions. Improper valve position (closed) would not affect the isolation function.

SR 3.1.8.2

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

SR 3.1.8.3

SR 3.1.8.3 is an integrated test of the SDV vent and drain valves to verify total system performance. After receipt of a simulated or actual scram signal, the closure of the SDV vent and drain valves is verified. The closure time of 30 seconds after receipt of a scram signal is based on the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.3 (continued)

bounding leakage case evaluated in the accident analysis (Ref. 3). Similarly, after receipt of a simulated or actual scram reset signal, the opening of the SDV vent and drain valves is verified. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1 and the scram time testing of control rods in LCO 3.1.3, "Control Rod OPERABILITY," overlap this Surveillance to provide complete testing of the assumed safety function. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 4.6.3.3.2.8.
2. 10 CFR 100.
3. NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," August 1981.

A.1

ITS 3.1.1

3.3 - LIMITING CONDITIONS FOR OPERATION

A. SHUTDOWN MARGIN (SDM)

LCO 3.1.1

The SHUTDOWN MARGIN (SDM) shall be equal to or greater than:

1. 0.38% Δk/k with the highest worth control rod analytically determined, or
2. 0.28% Δk/k with the highest worth control rod determined by test.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, 4, and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

ACTION A

1. In OPERATIONAL MODE 1 or 2, restore the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION B

A.2

ACTIONS C and D

2. In OPERATIONAL MODE 3 or 4, immediately verify all insertable control rods to be fully inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL MODE 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

A.3

A.5

ACTION D

A.4

3. In OPERATIONAL MODE 5, suspend CORE ALTERATION(s) and other activities that could reduce the SHUTDOWN MARGIN and fully insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

L.1

A.3

ACTION E

A.6

A.5

A.4

L.2

4.3 - SURVEILLANCE REQUIREMENTS

A. SHUTDOWN MARGIN

SR3.1.1.1

The SHUTDOWN MARGIN shall be determined to be equal to or greater than that specified at any time during the operating cycle:

LA.1

A.7

1. By demonstration, prior to or during the first startup after each refueling outage.

A.8

2. Within 24 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or known to be unscrammable. The required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or unscrammable control rod.

A.9 moved to ITS 3.1.3

LA.1

3. By calculation, prior to each fuel movement during the fuel loading sequence.

Page 1 of 1

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 In MODES 3 and 4, a single control rod may have been withdrawn under the provisions of the proposed LCO 3.10.2 and LCO 3.10.3, or some unanticipated event may have resulted in uninserted control rods. Therefore, rather than the passive CTS 3.3.A Action 2 words of "verify...inserted," the ITS 3.1.1 Required Actions C.1 and D.1 are active -- "Initiate action to fully insert..." This wording provides the same intent in the event all insertable control rods are inserted, and is therefore administrative.

A.3 CTS 3.3.A Actions 2 and 3 require suspension of activities that could reduce the SDM, when the SDM is not within limits in MODES 3, 4, or 5. In MODES 3 and 4, the vessel head is bolted in place; and the only activity that can significantly reduce SHUTDOWN MARGIN (SDM) is control rod withdrawal. Since a Required Action that ensures control rods remain inserted is provided, any additional action to suspend activities that can reduce the SDM is repetitive and unnecessary. Similarly, in MODE 5, the only activities that can affect SDM are CORE ALTERATIONS and control rod withdrawal. Since Required Actions are provided to suspend CORE ALTERATIONS and ensure control rods remain inserted, any additional action to suspend other activities is also repetitive and unnecessary. Therefore, these requirements in CTS 3.3.A Actions 2 and 3 have been deleted.

A.4 The CTS 3.3.A Actions 2 and 3 to "establish SECONDARY CONTAINMENT INTEGRITY within 8 hours" appear to provide a period of time (8 hours) in which integrity could be violated even if capable of being maintained. Additionally, if the plant status is such that integrity is not capable of being established within 8 hours, the existing Actions results in "non-compliance with the Technical Specifications" and a requirement for an LER. The intent of the Actions is more appropriately presented in ITS 3.1.1 Required Actions D.2, D.3, D.4, E.3, E.4, and E.5, which require actions to be initiated within one hour to restore the secondary containment boundary. With the proposed Required Actions, a significantly more conservative requirement to establish and maintain the secondary containment boundary is imposed. No longer would the

ADMINISTRATIVE

A.4 (cont'd) provision to violate the boundary for up to 8 hours exist. However, this conservatism comes from the understanding that if best efforts to establish the boundary exceeded 8 hours, no LER will be required.

This interpretation of the Actions intent is supported by the BWR ISTS, NUREG-1433, Revision 1. Because this is an enhanced presentation of existing intent, the proposed change is considered administrative.

A.5 This proposed change replaces the use of the defined term SECONDARY CONTAINMENT INTEGRITY in CTS 3.3.A Actions 2 and 3 with the essential elements of that definition. Refer also to the Discussion of Changes in the Definitions section (Chapter 1.0), which addresses deletion of the SECONDARY CONTAINMENT INTEGRITY definition. The change is editorial in that the requirements are specifically addressed by ITS 3.1.1 Required Actions D.2, D.3, D.4, E.3, E.4, and E.5. Therefore, the change is a presentation preference adopted by the BWR ISTS, NUREG-1433, Revision 1, and is considered administrative only.

A.6 The CTS 3.3.A Action 3 to "fully insert...within 1 hour" (see Discussion of Change L.2 below for the change to which control rods get inserted) is revised to "initiate action to fully insert...Immediately." This change is similar to that discussed in Discussion of Change A.4 above. The existing requirement appears to provide an hour in which control rods can be left withdrawn, even if able to be inserted. If the control rod is incapable of being inserted in 1 hour, the existing Action results in "non-compliance with the Technical Specifications" and a requirement for an LER. The intent of the Action is more appropriately presented in ITS 3.1.1 Required Action E.2. With the proposed Required Action, a significantly more conservative requirement to insert the control rod(s) and maintain insertion is imposed. No longer would the provision to withdraw or leave withdrawn one or more control rods for up to 1 hour exist. However, with this conservatism comes the understanding that if best efforts to insert the control rod(s) exceeds 1 hour, no LER will be required.

This interpretation of the Actions intent is supported by the BWR ISTS, NUREG-1433, Revision 1. Because this is an enhanced presentation of existing intent, the proposed change is considered administrative.

A.7 A specific completion time for the SDM test required by CTS 4.3.A.1 is proposed to clarify <u>when</u> "prior to or during the first startup" applies. Most SDM tests are performed as an in-sequence critical and, therefore, 4 hours after reaching criticality is provided in proposed SR 3.1.1.1 as a reasonable time to perform the required calculations and have appropriate verification completed.

ADMINISTRATIVE

A.7 (cont'd) Interpretations, both more and less conservative, can be made for the existing requirement; however, this interpretation of the Completion Time's intent is supported by the BWR ISTS, NUREG-1433, Revision 1. Because this is an enhanced presentation of existing intent, the proposed change is considered administrative.

A.8 More explicit wording is proposed to replace the activity referred to as "refueling outage" in CTS 4.3.A.1. The intent of the Surveillance Requirement is to perform the SDM test after in-vessel activities which could have altered SDM. These activities are explicitly stated in proposed SR 3.1.1.1 as "fuel movement within the reactor pressure vessel or control rod replacement." Because this is an enhanced presentation of the existing SR intent, the proposed change is considered administrative.

A.9 The CTS 4.3.A.2 requirement to perform an SDM test after finding a stuck control rod has been moved to ITS 3.1.3 in accordance with the BWR ISTS, NUREG-1433, Rev. 1. Any technical changes to this requirement will be discussed in the Discussion of Changes for ITS: 3.1.3.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details of the methods in CTS 4.3.A.1 (by demonstration) and CTS 4.3.A.3 (by calculation) to determine SHUTDOWN MARGIN (SDM) are proposed to be relocated to the Bases. The requirement in ITS SR 3.1.1.1 to verify SDM is within the specified limits is adequate to ensure that the requirement is met. Therefore, the relocated details are not required to be in the ITS to provide adequate protection to the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

"Specific"

L.1 The CTS 3.3.A Action 3 requirement to suspend all CORE ALTERATION(s) precludes off-loading fuel and inserting control rods. However, the insertion of control rods is allowed as indicated in the action but limited to within one hour after entry into the Condition. The one hour limitation has been changed as discussed in Discussion of Change A.6. The ITS 3.1.1 ACTION E modifies the requirement to suspend CORE ALTERATIONS "except for control rod insertion and fuel assembly removal." This exception allows continuation of activities that have a potential to correct the problem and restore a margin of safety to an inadvertent or uncontrolled core criticality. This additional operational flexibility does not require new or different actions, but allows corrective actions which would have otherwise been precluded (except under the provisions of 10 CFR 50.54(x)). The corrective actions would only be pursued in accordance with approved procedures.

L.2 The CTS 3.3.A Action 3 requirement to insert all insertable control rods in MODE 5 has been modified, ITS 3.1.1 Required Action E.2, to only require those control rods in core cells containing one or more fuel assemblies to be fully inserted. If all fuel assemblies are removed from a core cell, inserting the associated control rod has a negligible impact on core reactivity. During MODE 5, refueling procedures could have cells emptied and the control rod withdrawn, but "insertable." However, due to a variety of considerations (i.e., location of blade guides, ongoing instrumentation maintenance, water clarity), insertion of these control rods may not be desirable. Since there is negligible impact on SDM should the control rod be inserted with no fuel in the cell, it is acceptable to provide this flexibility.

RELOCATED SPECIFICATIONS

None

3.3 - LIMITING CONDITIONS FOR OPERATION

3.1.2 **B. Reactivity Anomalies** A.2

LCO 3.1.2

The reactivity equivalence of the difference between the actual critical control rod configuration and the predicted critical control rod configuration shall not exceed 1% Δk/k.

M.1

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

ACTION A With the reactivity equivalence difference exceeding 1% Δk/k, within ~~12~~ 72 hours perform an analysis to determine and explain the cause of the reactivity difference; operation may continue if the difference is explained and corrected. L.1 LA.1

ACTION B With the provisions of the ACTION above not met, be in at least HOT SHUTDOWN within the next 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

B. Reactivity Anomalies A.2

SR 3.1.2.1 The reactivity equivalence of the difference between the actual critical control rod configuration and the predicted critical control rod configuration shall be verified to be less than or equal to 1% Δk/k: A.3

1. During the first startup following CORE ALTERATION(s), and L.2
2. At least once per 31 effective full power days 1000 MWD/T L.3

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 The wording "reactivity equivalence of the difference" in CTS 3.3.B and CTS 4.3.B has been changed to "reactivity difference" to be consistent with NUREG-1433, Revision 1. This change does not affect the method utilized to verify this LCO, and therefore, the change is considered administrative.

A.3 A specific time for completing the reactivity anomaly surveillance CTS 4.3.B.1 is proposed to clarify when "during the first startup" the test must be performed. This test is performed by comparing the difference between the actual critical control rod configuration and the predicted critical control rod configuration as a function of cycle exposure while at steady state reactor power conditions. Therefore, "24 hours after reaching these conditions" is provided as a reasonable time to perform the required calculations and complete the appropriate verification. Interpretations, both more and less conservative, can be made for the existing requirement; however this interpretation of the intent is supported by the BWR ISTS, NUREG-1433, Revision 1. Because this is an enhanced presentation of existing intent, the proposed change is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS 3.3.B requires the reactivity difference between the actual critical control rod configuration and the predicted critical control rod configuration to be within limits. The CTS Bases clarifies that this verification can be performed by one of two methods: by comparison of the critical rod pattern selected base states to the predicted rod inventory at that state (i.e., rod density comparison) or by comparison of the monitored k_{eff} with the predicted k_{eff} as calculated by an approved 3-D core simulator code. These two methods to meet CTS 3.3.B were previously approved by the NRC in the SER for Amendment Nos. 177 and 175, dated May 23, 1997. Since Quad Cities 1 and 2 predict the core reactivity

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 (cont'd) using a 3-D simulator code and compare predicted k_{eff} with monitored k_{eff}, the alternate approach (i.e., the control rod density comparison) is not necessary. Therefore, ITS 3.1.2 will explicitly require the comparison between monitored and predicted k_{eff}. Since the alternate approach has been deleted, this change is considered more restrictive on plant operation. However, the proposed requirement in ITS 3.1.2 continues to be adequate to ensure the safety analysis is met.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The requirement of CTS 3.3.B Action to perform an analysis to determine and explain the cause of the reactivity difference is proposed to be relocated to the Bases. This requirement involves re-evaluating predicted core reactivity conditions in an effort to explain and correct the difference such that, based on the new evaluation, the reactivity difference is returned to acceptable limits. The action to restore compliance to within the limit is maintained in Required Action A.1. As a result, these details associated with the method of restoring compliance to within the limit are not necessary to ensure restoration is accomplished in a timely manner. Therefore, the relocated requirement is not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the Proposed Bases Control Program described in Chapter 5 of the ITS.

"Specific"

L.1 The time allowed to restore the core reactivity difference to within limits in the CTS 3.3.B Action (i.e., to "perform an analysis to determine and explain the cause of the reactivity difference") has been increased from 12 hours to 72 hours. 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 a fuel analysis department and the fuel vendor. Contacting 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 ISTS, NUREG-1433, Revision 1 allows this time to be extended to 72 hours.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

L.2 The term "CORE ALTERATION(s)" in CTS 4.3.B.1 is proposed to be replaced with "fuel movement within the reactor pressure vessel or control rod replacement." The intent of this Surveillance is to verify the core reactivity after in-vessel operations which could have significantly altered the core reactivity. Certain CORE ALTERATIONS have a known effect which is reversible and, are consistent with the activities assumed to occur during routine operations. Normal control rod movement is such an activity. Since this activity does not require reverification of core reactivity during normal operations with the vessel head on (i.e., not defined as a CORE ALTERATION), it should also be allowed without a requirement to reverify core reactivity, with the reactor vessel head removed (i.e., defined as a CORE ALTERATION). The proposed wording provides a specific list of those CORE ALTERATIONS which constitute a core reactivity change not expected to occur during normal operations, specifically excluding normal control rod movement.

L.3 The frequency in CTS 4.3.B.2, "31 effective full power days" (approximately 625 MWD/T), is proposed to be replaced with "1000 MWD/T during operations in MODE 1." Both Frequencies consider the relatively slow change in core reactivity with exposure and operating experience related to variations in core reactivity. The proposed change is consistent with the BWR ISTS, NUREG-1433, Revision 1.

RELOCATED SPECIFICATIONS

None

A.1

A.2

3.3 - LIMITING CONDITIONS FOR OPERATION

C. Control Rod OPERABILITY

LCO 3.1.3 All control rods shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

A.3 A.4

Add proposed ACTIONS Note

Add proposed Required Action A.1 Note

ACTION:

ACTION A 1. With one control rod inoperable due to being ~~immovable as a result of excessive friction or mechanical interference, or known to be unscrammable:~~ A.5

M.1 Add proposed Required Action A.1

L.1 ACTION D a. Within ~~one~~ hour:

1) Verify that the inoperable control rod ~~, if withdrawn,~~ is separated from all other inoperable ~~withdrawn~~ control rods by at least two control cells in all directions. M.2

L.2 Required Action A.2 2) Disarm the associated ~~directional control valves~~[a] ~~either:~~ control rod drive (CRD)

~~a) Electrically, or~~ M.3

~~b) Hydraulically by closing the drive water and exhaust water isolation valves.~~ LA.1

Add proposed ACTION B M.3

ACTION F b. With the provisions of ACTION 1.a above not met, be in at least HOT SHUTDOWN within the next 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

C. Control Rod OPERABILITY

SR 3.1.3.2 SR 3.1.3.3 1. When above the low power setpoint of the RWM, all withdrawn control rods ~~not required to have their directional control valves disarmed electrically or hydraulically~~ shall be demonstrated OPERABLE by ~~moving~~ each control rod at least one notch: A.6 M.5

SR 3.1.3.2 Note

SR 3.1.3.2 a. At least once per 7 days[b] for each fully withdrawn control rod, and at

SR 3.1.3.3 least once per 31 days[b] for each partially withdrawn control rod, and

SR 3.1.3.3 Note

Required Action A.3 b. Within 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference, or known to be unscrammable.

A.7

~~2. All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.3.D, 4.3.F, 4.3.G, 4.3.H and 4.3.I.~~

A.8

~~a May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.~~

SR 3.1.3.2 and SR 3.1.3.3 b Not required to be performed until 7 days (for fully withdrawn) or 31 days (for partially withdrawn) after the control rod is withdrawn and above the low power setpoint of the RWM.

A.1

ITS 3.1.3

3.3 - LIMITING CONDITIONS FOR OPERATION

4.3 - SURVEILLANCE REQUIREMENTS

Required Action A.4

c. Comply with Surveillance Requirement 4.3.A.2 within 24 hours or be in HOT SHUTDOWN within the next 12 hours.

72 L.4

ACTION F

ACTION C

2. With one or more control rods scrammable but inoperable for causes other than addressed in ACTION 3.3.C.1 above:

add proposed Note to Condition D L.1

a. If the inoperable control rod(s) is withdrawn, within one hour:

L.1

ACTION D

1) Verify that the inoperable ~~withdrawn~~ control rod(s) is separated from all other inoperable ~~withdrawn~~ control rods by at least two control cells in all directions, and

M.2

A.4

add proposed Required Action C.1 Note

L.5

Required Actions C.1 and C.2

2) Demonstrate the insertion capability of the inoperable withdrawn control rod(s) by inserting the inoperable withdrawn control rod(s) at least one notch by drive water pressure within the normal operating range.[b]

M.6

b. With the provisions of ACTION 2.a above not met, fully insert the inoperable withdrawn control rod(s) and disarm the associated directional control valves[a] either:

CRD LA.1

1) ~~Electrically, or~~

2) ~~Hydraulically by closing the drive water and exhaust water isolation valves.~~

M.6

b The inoperable control rod may then be withdrawn to a position no further withdrawn than its position when found to be inoperable.

A.8

a ~~May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.~~

3.3 - LIMITING CONDITIONS FOR OPERATION

4.3 - SURVEILLANCE REQUIREMENTS

Required Action C.2

c. If the inoperable control rod(s) is fully inserted, within one hour disarm the associated directional control valves[a] either:

4 L.5

CRD LA.1

1) Electrically, or

2) Hydraulically by closing the drive water and exhaust water isolation valves.

ACTION F 3. With the provisions of ACTION 2 above not met, be in at least HOT SHUTDOWN within the next 12 hours.

ACTION F 4. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

A.8

a May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

A. SHUTDOWN MARGIN (SDM)

The SHUTDOWN MARGIN (SDM) shall be equal to or greater than:

1. 0.38% Δk/k with the highest worth control rod analytically determined, or

2. 0.28% Δk/k with the highest worth control rod determined by test.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, 4, and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

1. In OPERATIONAL MODE 1 or 2, restore the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.

2. In OPERATIONAL MODE 3 or 4, immediately verify all insertable control rods to be fully inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL MODE 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

3. In OPERATIONAL MODE 5, suspend CORE ALTERATION(s) and other activities that could reduce the SHUTDOWN MARGIN and fully insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

4.3 - SURVEILLANCE REQUIREMENTS

A. SHUTDOWN MARGIN

Required Action A.4

The SHUTDOWN MARGIN shall be determined to be equal to or greater than that specified at any time during the operating cycle:

1. By demonstration, prior to or during the first startup after each refueling outage.

Required Action A.4

2. Within ~~24~~ 72 L.4 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or known to be unscrammable. A.5 The required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or unscrammable control rod. A.9 moved to ITS Chapter 1.0 SDM definition

3. By calculation, prior to each fuel movement during the fuel loading sequence.

See ITS 3.1.1

general organization A.10

3.3 - LIMITING CONDITIONS FOR OPERATION

D. Maximum Scram Insertion Times

SR 3.1.3.4 The maximum scram insertion time of each control rod from the fully withdrawn position to 90% insertion, ~~based on de-energization of the scram pilot valve solenoids as time zero,~~ A.11 shall not exceed 7 seconds.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

ACTION A or ACTION C With the maximum scram insertion time of one or more control rods exceeding 7 seconds:

1. Declare the control rod(s) exceeding the above maximum scram insertion time inoperable, and

L.6

2. ~~When operation is continued with three or more control rods with maximum scram insertion times in excess of 7 seconds, perform Surveillance Requirement 4.3.D.3 at least once per 60 days of POWER OPERATION.~~

~~With the provisions of the ACTION above not met, be in at least HOT SHUTDOWN within 12 hours.~~

4.3 - SURVEILLANCE REQUIREMENTS

D. Maximum Scram Insertion Times

The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than 800 psig and, during single control rod scram time tests, with the control rod drive pumps isolated from the accumulators:

1. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER:

   a. following CORE ALTERATION(s), or

   b. after a reactor shutdown that is greater than 120 days,

2. For specifically affected individual control rods[a] following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and

3. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of POWER OPERATION.

A.12 add proposed SR 3.1.3.4

see ITS 3.1.4

a The provisions of Specification 4.0.D are not applicable provided this surveillance is conducted prior to exceeding 40% of RATED THERMAL POWER.

3.3 - LIMITING CONDITIONS FOR OPERATION

SR 3.1.3.5

H. Control Rod Drive Coupling

All control rods shall be coupled to their drive mechanisms.

A.13

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, ~~and 5[a]~~.

L.7

ACTION:

1. In OPERATIONAL MODE 1 or 2 with one control rod not coupled to its associated drive mechanism, within 2 hours:

L.5

L.8

a. ~~If permitted by the RWM,~~ ~~insert the control rod drive mechanism to accomplish recoupling~~ ~~and verify recoupling by withdrawing the control rod, and:~~

A.14

L.9

L.10

~~1) Observing any indicated response of the nuclear instrumentation, and~~

~~2) Demonstrating that the control rod will not go to the overtravel position.~~

L.8

ACTION C

b. ~~If not permitted by the RWM or, if recoupling is not accomplished in accordance with ACTION 1.a above,~~ then declare the control rod inoperable, fully insert the control rod and disarm the associated CRD ~~directional control valves[b] either:~~

LA.1

~~1) Electrically, or~~

4.3 - SURVEILLANCE REQUIREMENTS

H. Control Rod Drive Coupling

SR 3.1.3.5 Each affected control rod shall be demonstrated to be coupled to its drive mechanism by verifying that the control rod drive does not go to the overtravel position:

~~1. Deleted.~~

2. Anytime the control rod is withdrawn to the "Full out" position, and

3. Following maintenance on or modification to the control rod or control rod drive system which could have affected the control rod drive coupling integrity.

L.7

~~a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.~~

A.8

~~b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.~~

3.3 - LIMITING CONDITIONS FOR OPERATION

4.3 - SURVEILLANCE REQUIREMENTS

2) Hydraulically by closing the drive water and exhaust water isolation valves.

LA.1

ACTION F

2. With the provisions of ACTION 1 above not met, be in at least HOT SHUTDOWN within 12 hours.

3. In OPERATIONAL MODE 5[a] with a withdrawn control rod not coupled to its associated drive mechanism, within 2 hours:

a. Insert the control rod to accomplish recoupling and verify recoupling by withdrawing control rod and demonstrating that the control rod will not go to the overtravel position, or

b. If recoupling is not accomplished, declare the control rod inoperable, fully insert the control rod and disarm the associated directional control valves[b] within one hour, either:

1) Electrically, or

2) Hydraulically by closing the drive water and exhaust water isolation valves.

L.7

L.7

a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3/10.I or 3.10.J.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

L.7

3.3 - LIMITING CONDITIONS FOR OPERATION

I. Control Rod Position Indication System

SR 3.1.3.1 All control rod position indicators shall be OPERABLE.

A.15

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, ~~and 5[a].~~

moved to ITS 3.9.4 A.16

ACTION:

ACTION C 1. In OPERATIONAL MODE 1 or 2 with one or more control rod position indicators inoperable, within one hour either: L.5

a. Determine the position of the control rod ~~by an alternate method, or~~ LA.2

~~b. Move the control rod to a position with an OPERABLE position indicator, or~~ LA.2

c. Declare the control rod inoperable, fully insert the inoperable withdrawn control rod(s), and disarm the ~~associated directional control valves[b] either:~~ CRD

~~1) Electrically, or~~

~~2) Hydraulically by closing the drive water and exhaust water isolation valves.~~ LA.1

4.3 - SURVEILLANCE REQUIREMENTS

I. Control Rod Position Indication System

SR 3.1.3.1 The control rod position indication system shall be determined OPERABLE by verifying:

1. At least once per 24 hours that the position of each control rod is indicated.

2. ~~That the indicated control rod position changes during the movement of the control rod drive when performing Surveillance Requirement 4.3.C.1.~~ L.11

3. Deleted.

A.16 moved to ITS 3.9.4

a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

b ~~May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod(s) to OPERABLE status.~~ A.8

3.3 - LIMITING CONDITIONS FOR OPERATION

4.3 - SURVEILLANCE REQUIREMENTS

ACTION F

2. With the provisions of ACTION 1 above not met, be in at least HOT SHUTDOWN within the next 12 hours.

3. In OPERATIONAL MODE 5[a] with a withdrawn control rod position indicator inoperable:

A.16 moved to ITS 3.9.4

a. Move the control rod to a position with an OPERABLE position indicator, or

b. Fully insert the control rod.

A.16 moved to ITS 3.9.4

a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 The organization of the Control Rod OPERABILITY Specification (ITS 3.1.3) is proposed to include all conditions that can affect the ability of the control rods to provide the necessary reactivity insertion. The proposed Specification is also simplified as follows:

1) A control rod is considered "inoperable" only when it is degraded to the point that it cannot provide its scram functions (i.e., scram insertion times, coupling integrity, and ability to determine position). 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. Requirements are retained to preserve SHUTDOWN MARGIN for this situation.

3) Special considerations are provided for nonconformance to the analyzed rod position due to inoperable control rods, at < 10% of RATED THERMAL POWER.

A.3 A proposed ACTIONS Note, "Separate Condition entry is allowed for each control rod," has been added to CTS 3.3.C Actions (ITS 3.1.3 ACTIONS) and provides 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 existing ACTIONS for inoperable control rods. It is intended that each inoperable control rod is allowed a specified period of time in which compliance with certain limits is verified and, when necessary, the control rod is fully inserted and disarmed.

A.4 A Note is added to CTS 3.3.C, Actions 1 and 2 (ITS 3.1.3 Required Actions Notes A.1 and C.1) that allows for bypassing the RWM, if needed for continued operations. This note is informative in that the RWM may be bypassed at any

ADMINISTRATIVE

A.4 (cont'd) time, provided the proper ACTIONS of CTS 3.3.L (ITS 3.3.2.1), the RWM Specification, are taken. This is a human factors consideration to assure clarity of the requirement and allowance.

A.5 The existing phrase of "Immovable, as a result of excessive friction or mechanical interference, or known to be unscrammable" in CTS 3.3.C Action 1 and CTS 4.3.A.2 has been replaced with the term "stuck" in proposed Condition A of ITS 3.1.3. The intent of the existing wording is consistent with the proposed simplification. Details of potential mechanisms by which control rods may be stuck are not necessary for inclusion within the Condition.

A.6 CTS 4.3.C.1 pertains to control rods "not required to have their directional control valves disarmed electrically or hydraulically." This phrase thus exempts this surveillance for inoperable control rods. Currently, inoperable control rods are already not required to meet this Surveillance (per CTS 4.0.D), and therefore, CTS 4.3.C.1 only applies to OPERABLE control rods. Therefore, this phrase is proposed to be deleted since it is not needed.

A.7 These listed Surveillances in CTS 4.3.C.2 are required by other Specifications. Repeating a requirement to perform these Surveillances is not necessary. Elimination of this "cross-reference" is therefore administrative.

A.8 CTS 3.3.C Actions 1.a.2), 2.b, and 2.c footnote a, CTS 3.3.H Action 1.b footnote b, and CTS 3.3.I Action 1.c footnote b, which permit the directional control valves to be rearmed intermittently, have been deleted since proposed LCO 3.0.5 provides this allowance (i.e., this allowance has been moved to LCO 3.0.5). Therefore, deletion of this allowance is administrative.

A.9 The SDM allowance in CTS 4.3.A.2 is being moved to the definition of SDM in proposed Section 1.1, in accordance with the BWR ISTS, NUREG-1433, Revision 1. Any technical changes to these requirements will be discussed in the Discussion of Changes for ITS: Chapter 1.0.

A.10 The CTS 3.3.D requirement that maximum control rod scram insertion time be ≤ 7 seconds is presented in proposed SR 3.1.3.4, making it a requirement for control rods to be considered OPERABLE. Eliminating the separate Specification for excessive scram time by moving the requirement to a Surveillance Requirement does not eliminate any of the requirements, or impose a new or different treatment of the requirements (other than those proposed in Discussion of Change L.6 below). Therefore, this proposed change is administrative.

ADMINISTRATIVE (continued)

A.11 The definition of time zero in CTS 3.3.D (i.e., "based on de-energization of the scram pilot valve solenoids as time zero") has been deleted since it is duplicative of the definition of time zero in CTS 3.3.E and 3.3.F, which is maintained in proposed footnote (a) to ITS Table 3.1.4-1. No change has been made to the defined time zero, therefore, this deletion is administrative.

A.12 CTS 4.3.D, which provides the scram time testing requirements, is addressed in ITS 3.1.4. Therefore, proposed SR 3.1.3.4 has been added to require the SRs in ITS 3.1.4 to be performed. Changes to the testing requirements located in LCO 3.1.4 as SRs 3.1.4.1, 3.1.4.2, 3.1.4.3, and 3.1.4.4 are addressed in the Discussion of Changes for ITS: 3.1.4.

A.13 The CTS 3.3.H requirement that control rods be coupled to their drive mechanism is presented in proposed SR 3.1.3.5. As a Surveillance in the Control Rod OPERABILITY LCO, it is a requirement for control rods to be considered OPERABLE. The actions for uncoupled control rods continue to be required (see Discussion of Changes L.5, L.7, L.8, L.9, and L.10 below). Eliminating the separate LCO for control rod coupling, by moving the Surveillance and ACTIONS to another Specification, 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 administrative.

A.14 CTS 3.3.H Action 1.a contains the method of restoring coupling integrity to an uncoupled control rod (insert the control rod drive mechanism to accomplish recoupling). The revised presentation of actions (based on the BWR ISTS, NUREG-1433, Revision 1) is proposed to not explicitly detail options to "restore...to OPERABLE." This action is always an option, and is implied in all ACTIONS. Omitting this action is purely editorial.

A.15 CTS 3.3.I requires all control rod position indicators to be Operable. The intent of the CTS 3.3.I requirement is understood to be related to each control rod. Each specific action within Action 1, Action 3, and each Surveillance Requirement all refer to individual control rods. Therefore, the interpretation of this LCO is that each control rod shall have "at least one control rod position indication."

The essence of the requirement that each control rod have at least one control rod position indication is presented in SR 3.1.3.1 of ITS 3.1.3, "Control Rod OPERABILITY." The effect of relocating the requirement for control rod position indication is to make it a requirement for control rods to be considered OPERABLE. Eliminating the separate LCO for control rod position indication

ADMINISTRATIVE

A.15 (cont'd) (by moving the Surveillance and ACTIONS to another Specification) does not eliminate any requirements or impose a new or different treatment of the requirements (other than those separately proposed). Similarly, CTS 3.3.I Action 1 addresses this intent. The proposed SR 3.1.3.1 has combined the CTS 3.3.I intent with the CTS 3.3.I Action 1 intent to require the position of the control rod be determined. If the position can be determined, the control rod may be considered OPERABLE, and continued operation allowed. This outcome is identical, whether complying with CTS 3.3.I Action 1, or meeting proposed SR 3.1.3.1.

A.16 The CTS 3.3.I requirements, including Action 3, for control rod position indication during refueling (OPERATIONAL MODE 5) are being moved to Section 3.9 in accordance with the format of the BWR ISTS, NUREG-1433, Revision 1. Any technical changes to these requirements will be discussed in the Discussion of Changes for ITS: 3.9.4.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 A proposed Required Action has been added to CTS 3.3.C Action 1.a.1) for a stuck control rod. ITS 3.1.3 Required Action A.1 requires the immediate verification that the stuck control rod separation criteria are met. The actual criteria are specified in the Bases and are applicable to both GE and Siemens fuel. The stuck control rod separation criteria are not met if: a) the stuck control rod occupies a location adjacent to two "slow" control rods, b) stuck control rod occupies a location adjacent to one "slow" control rod, and the one "slow" control rod is also adjacent to another "slow" control rod, or c) if the stuck control rod occupies a location adjacent to one "slow" control rod when there is another pair of "slow" control rods elsewhere in the core adjacent to one another. The description of "slow" control rods is provided in LCO 3.1.4, "Control Rod Scram Times" (see Discussion of Changes for ITS 3.1.1 in this section). The stuck separation criteria ensures local scram reactivity rate assumptions are met.

M.2 CTS 3.3.C Actions 1.a.1) and 2.a.1) require the separation criteria to be met only for withdrawn control rods. Condition D of the ITS 3.1.3 applies to all inoperable control rods (when $\leq 10\%$ RTP, see Discussion of Change L.1 below) whether inserted or withdrawn, and is therefore, more restrictive. This revised separation criteria requirement is necessary to ensure the safety analysis assumptions are met.

TECHNICAL CHANGES - MORE RESTRICTIVE (continued)

M.3 The CTS 3.3.C Actions require LCO 3.0.C (ITS LCO 3.0.3) entry if more than one control rod is stuck. The proposed ITS 3.1.3 ACTION B maintains the equivalent shutdown action as LCO 3.0.3, but also contains an additional requirement to disarm the stuck control rod (ITS 3.1.3 Required Action A.2). The Bases for this Required Action requires the disarming to be performed hydraulically. This additional requirement provides a necessary level of protection to the control rod drive should a scram signal occur. If mechanically bound, the stuck control rod could cause further damage if not hydraulically disarmed. Disarming normally would preclude control rod insertion on a scram signal; however, since this control rod is stuck, this effect of disarming is moot. In addition, CTS 3.3.C Action 1.a.2)a) allows a stuck control rod to be disarmed electrically. This allowance has been deleted. The stuck control rod can only be disarmed hydraulically. This will also prevent potential damage if a scram signal occurs, since the means by which hydraulic disarming is performed will preclude scram pressure from being applied.

M.4 Not used.

M.5 Proposed SR 3.1.3.2 and SR 3.1.3.3 require control rods to be inserted in lieu of the CTS 4.3.C.1 requirement for "moving." The existing requirement can be met by control rod withdrawal. It is conceivable that a mechanism causing binding of the control rod that prevents insertion can exist such that a withdrawal test will 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.

M.6 The proposed changes to CTS 3.3.C Action 2.a.2) including footnote (b), for non-stuck inoperable control rods, eliminates the check of insertion capability; replacing it with a requirement to fully insert and disarm all inoperable control rods. CTS 3.3.C Action 2.a.2), requiring the insertion capability to be verified and allowing the control rod to remain withdrawn, is applicable to conditions such as: 1) one inoperable CRD accumulator, and 2) loss of position indication while below the low power setpoint. The first condition is addressed in the Discussion of Changes for ITS: 3.1.5. The latter condition would no longer allow the affected control rod to remain withdrawn and not disarmed. This added restriction on control rod(s) with loss of position indication is conservative with respect to scram time and SDM since an inoperable (but not stuck) control rod is not disarmed while it is withdrawn. ACTIONS for inoperable control rods not complying with analyzed rod position sequence (ITS 3.1.3 ACTION D) assure that insertion of these control rods remain appropriately controlled.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details of the recommended procedures for disarming control rod drives (CRDs) specified in CTS 3.3.C Actions 1.a.2) (with the exception of electrical disarming, see Discussion of Change M.3 above), 2.b, and 2.c, CTS 3.3.H Action 1.b, and CTS 3.3.I Action 1.c are proposed to be relocated to the Bases. These details are not necessary to ensure the associated CRDs of inoperable control rods are disarmed. ITS 3.1.3 Required Actions A.2 and C.2, which require disarming the associated CRDs of inoperable control rods, are adequate for ensuring associated CRDs and inoperable control rods are disarmed. Therefore, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

LA.2 CTS 3.3.I Actions 1.a and 1.b, which determine the position of the control rod (now proposed to be a Surveillance for control rod OPERABILITY - refer to Discussion of Change A.15 above) can be met a number of ways. Two ways are presented: by using an alternate method and by moving the control rod to a position with an OPERABLE position indicator. These details of methods for determining the position of a control rod are proposed to be relocated to the Bases for the proposed Surveillance (SR 3.1.3.1). SR 3.1.3.1, which requires the position of each control rod to be determined every 24 hours, is adequate for ensuring the position of the control rods is determined. Therefore, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

"Specific"

L.1 CTS 3.3.C Actions 1.a and 2.a are presented in ITS 3.1.3 ACTION D to provide the requirements and actions for the local distribution of inoperable control rods. Three distinct changes are addressed:

1) ITS 3.1.3 ACTION D is modified by a Note excluding its applicability above 10% power. The existing separation requirements for a stuck control rod, in part, account for allowing withdrawn inoperable control rods. (See Discussion of Change M.2 above.) To preserve scram reactivity, a stuck rod must be separated from other withdrawn inoperable control rods which may also not scram. In the ITS, all inoperable control rods which will not scram or cannot be verified to

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 (continued)

scram (e.g., loss of position indication) are required to be fully inserted, and therefore, cannot impact scram reactivity. Therefore, scram reactivity remains preserved at all power levels and is unaffected by this proposed change. Separation requirements are required when below 10% power because of Control Rod Drop Accident (CRDA) concerns related to control rod worth. Above 10% power, control rod worths that are of concern for the CRDA are not possible.

2) ITS 3.1.3 ACTION D also does not require actions for inoperable control rods whose position is in conformance with the analyzed rod position sequence (e.g., BPWS) constraints, even if the inoperable control rods are within two cells of each other. As discussed above in the first item of this change, adequate limits to control core reactivity and power distribution above 10% power remain with this proposed change. Below 10% power, the appropriate core reactivity and power distribution limits are controlled by maintaining control rod positions within the limits of the analyzed rod position sequence and maintaining scram times within the limits of CTS 3.3.E and 3.3.F (as modified to reflect ITS 3.1.4). If the two inoperable control rods were both "stuck," Required Actions require an immediate shutdown, regardless of their proximity. Therefore, the limitation on the local distribution of inoperable control rods that comply with the analyzed rod position sequence is overly restrictive.

3) Finally, the Required Actions for ITS 3.1.3 ACTION D allow 4 hours to correct the situation prior to commencing a required shutdown, while CTS 3.3.C Actions 1.a and 2.a allow 1 hour. This increase is proposed in recognition of the actual operational steps involved on discovery of inoperable control rod(s). Time is first required to attempt identification and correction of the problem. Additional time is necessary to fully insert (some operational considerations may be necessary to adjust control rod patterns and/or power levels), and then disarm the affected control rod(s). After these high priority steps are accomplished, attention can be turned to correcting localized distribution of inoperable control rods that deviate from the analyzed rod position sequence. Given the low probability of a CRDA during this brief proposed time extension, and the desire not to impose excessive time constraints on operator actions that could lead to hasty corrective actions, the proposed extension to this action does not represent a significant safety concern.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

L.2 Disarming a control rod as required by CTS 3.3.C Action 1.a.2) involves personnel actions by other than control room operating personnel. These processes require coordination of personnel and preparation of equipment, and potentially require anti-contamination "dress-out," in addition to the actual procedure of disarming the control rod. Currently, all these activities must be completed and the control room personnel must confirm completion within the same 1 hour allowed to insert the control rod. This is proposed to be extended to 2 hours in ITS 3.1.3 Required Action A.2 (consistent with the BWR ISTS, NUREG-1433, Revision 1) in recognition of the potential for excessive haste required to complete this task. The proposed 2 hour time does not represent a significant safety concern as the control rod is already in an acceptable position (in accordance with other ACTIONS), and the ACTION to disarm is solely a mechanism for precluding the potential for damage to the CRD mechanism.

L.3 Not used.

L.4 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 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 this postulated additional single failure, sufficient reactivity control remains to reach and maintain HOT SHUTDOWN conditions. Also, a notch test is required by ITS 3.1.3 Required Action A.3 for each remaining withdrawn control rod to ensure that no additional control rods are stuck. Given these considerations, the time to demonstrate SHUTDOWN MARGIN in CTS 3.3.C Action 1.c and CTS 4.3.A.2 has been extended from 24 hours to 72 hours, and provides a reasonable time to perform the analysis or test.

L.5 CTS 3.3.C Action 2 (for excessive scram speed, certain combinations of conditions with a low pressure on a control rod scram accumulator), CTS 3.3.H Action 1 (for uncoupled control rods), and CTS 3.3.I Action 1 (for inoperable control rod position indication) provide actions for inoperable control rods. Both CTS 3.3.C Action 2 and CTS 3.3.H Action 1 provide a total of two hours to insert and disarm the control rods while CTS 3.3.I provides only one hour. In the ITS all inoperable non-stuck control rods are required to be fully inserted and disarmed (see Discussion of Changes M.6 above). The time allowed to complete the insertion is proposed to be extended to 3 hours (ITS 3.1.3 Required Action C.1) for all cases an additional hour is provided to disarm the associated CRD (ITS 3.1.3 Required Action C.2). The additional time provided to disarm the

TECHNICAL CHANGES - LESS RESTRICTIVE

L.5 (cont'd) associated CRD (ITS 3.1.3 Required Action C.2). The additional time provides the necessary time to insert and disarm the control rods in an orderly manner and without challenging plant systems. The Rod Worth Minimizer may be required to by bypassed to allow the rod to be inserted, therefore, the current action times may not be sufficient under all cases.

In addition, disarming a control rod can involve personnel actions by other than control room operating personnel. This process requires coordination of personnel and preparation of equipment, and potentially requires anti-contamination "dress-out," in addition to the actual procedure of disarming the control rod.

The disarming is proposed to be extended to 4 hours in ITS 3.1.3 Required Action C.2, 1 hour beyond that allowed to insert (consistent with the BWR ISTS, NUREG-1433, Revision 1) in recognition of the potential for excessive haste required to complete this task. The proposed 4 hour time does not represent a significant safety concern since the control rod will be inserted within 3 hours and the action to disarm is solely a mechanism for precluding the potential for future misoperation.

L.6 The CTS 3.3.D Action 2 requirement for additional scram time surveillance testing when three or more control rods exceed the maximum scram time is deleted. During normal power operating conditions, scram testing is a significant perturbation to steady state operation, involving significant power reductions, abnormal control rod patterns and abnormal control rod drive hydraulic system configurations. Requiring more frequent scram time surveillance tests is therefore not desirable. Because of the frequent testing of control rod insertion capability (proposed SR 3.1.3.2 and SR 3.1.3.3) and accumulator OPERABILITY (proposed SR 3.1.5.1), and the operating history demonstrating a high degree of reliability, the more frequent scram time testing is not necessary to assure safe plant operations. In addition, since the shutdown requirement ("with the provisions of the ACTION above not met, be in at least HOT SHUTDOWN within 12 hours") could have only applied to CTS 3.3.D Action 2 (since a control rod can always be declared inoperable), this part of CTS 3.3.D Action 2 has also been deleted.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

L.7 Coupling requirements during refueling (OPERATIONAL MODE 5) specified by CTS 3/4.3.H are not necessary since only one control rod can be withdrawn from core cells containing fuel assemblies. The probability and consequences of a single control rod dropping from its fully inserted position to the withdrawn position of the control rod drive are negligible (i.e., reactor will remain subcritical and within the limits of the CRDA assumptions). However, these requirements are retained for the proposed SDM testing in MODE 5 (ITS 3.10.7).

L.8 If an uncoupled control rod is not allowed by the RWM to be inserted to accomplish recoupling, CTS 3.3.H Action 1.b requires the control rod be inserted. This will require bypassing the RWM and operation with an out-of-sequence control rod. Therefore, coupling attempts are allowed regardless of the RWM allowance because of the short time allowed. If coupling is not established within 3 hours, the control rod must be fully inserted and disarmed (ITS 3.1.3 Required Actions C.1 and C.2).

L.9 Proposed SR 3.1.3.5 verifies a control rod does not go to the withdrawn overtravel position. An uncoupled control rod would fail to meet SR 3.1.3.5. After restoration of a component that caused a required SR to be failed, SR 3.0.1 requires the appropriate SRs (in this case SR 3.1.3.5) to be performed to demonstrate the OPERABILITY of the affected components. The requirement to verify control rod coupling by observation of nuclear instrumentation response is addressed in Discussion of Change L.10 below. As a result, the CTS 3.3.H Actions 1.a and 1.a.2) requirements are proposed to be deleted since they are not necessary for ensuring recoupling of the control rod.

L.10 The CTS 3.3.H Action 1.a.1) requirement to verify control rod coupling by observing any indicated response of the nuclear instrumentation during withdrawal of a control rod are proposed to be deleted. A response to control rod motion on nuclear instrumentation is indicative that a control rod is following its drive, but gives no indication as to whether or not a control rod is coupled. Likewise, failure to have a response to control rod motion on nuclear instrumentation does not indicate that a rod is uncoupled. Thus, the results from monitoring nuclear instrumentation are inconclusive to use as a verification that the control rod is coupled. Proposed SR 3.1.3.5 requires verification that a control rod does not go to the withdrawn overtravel position. The overtravel feature provides a positive check of coupling integrity since only an uncoupled control rod can go to the overtravel position. This verification is required to be

TECHNICAL CHANGES - LESS RESTRICTIVE

L.10 (cont'd) performed any time a control rod is withdrawn to the full out position and prior to declaring a control rod operable after work on the control rod or Control Rod Drive System that could affect coupling. As a result, SR 3.1.3.5 provides adequate assurance that the control rods are coupled.

L.11 CTS 4.3.I.2 requires that the indicated control rod position change during the movement of the control rod drive when performing the control rod movement tests (CTS 4.3.C.1). To perform control rod movement tests required by CTS 4.3.C.1 (proposed SR 3.1.3.2 and SR 3.1.3.3), position indication must be available. If position indication is not available, this test cannot be satisfied and appropriate actions will be taken for inoperable control rods in accordance with the ACTIONS of ITS 3.1.3. As a result, the requirements for the control rod position indication system are adequately addressed by the requirements of ITS 3.1.3 and associated SR 3.1.3.2 and SR 3.1.3.3 and are proposed to be deleted.

RELOCATED SPECIFICATIONS

None

3.3 - LIMITING CONDITIONS FOR OPERATION

D. Maximum Scram Insertion Times

The maximum scram insertion time of each control rod from the fully withdrawn position to 90% insertion, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7 seconds.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding 7 seconds:

1. Declare the control rod(s) exceeding the above maximum scram insertion time inoperable, and
2. When operation is continued with three or more control rods with maximum scram insertion times in excess of 7 seconds, perform Surveillance Requirement 4.3.D.3 at least once per 60 days of POWER OPERATION.

With the provisions of the ACTION above not met, be in at least HOT SHUTDOWN within 12 hours.

see ITS 3.1.3

4.3 - SURVEILLANCE REQUIREMENTS

M.1

D. Maximum Scram Insertion Times

or equal to

SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.4

The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than 800 psig ~~and, during single~~ control rod scram time tests, with the control rod drive pumps isolated from the accumulators: NOTE to Surveillance Requirements

1. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER:

L.1

SR 3.1.4.4 a. following CORE ALTERATION(s), or

SR 3.1.4.1 b. after a reactor shutdown that is greater than 120 days,

add proposed SR 3.1.4.3

M.1

SR 3.1.4.4 2. For specifically affected individual control rods[a] following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and

SR 3.1.4.2

3. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of POWER OPERATION.

LA.1

A.2

SR 3.1.4.4 a The provisions of Specification 4.0.D are not applicable provided this surveillance is conducted prior to exceeding 40% of RATED THERMAL POWER.

3.3 - LIMITING CONDITIONS FOR OPERATION

E. Average Scram Insertion Times

Footnote (a) to Table 3.1.4-1

The average scram insertion time of all OPERABLE control rods from the fully withdrawn position, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.900
50	2.00
90	3.50

M.2

add proposed LCO 3.1.4 and Table 3.1.4-1

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

ACTION A

With the average scram insertion time exceeding any of the above limits, be in at least HOT SHUTDOWN within 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

E. Average Scram Insertion Times

The control rod average scram times shall be demonstrated by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.3.D.

SR 3.1.4.1, SR 3.1.4.2 and SR 3.1.4.4

ITS 3.1.4

3.3 - LIMITING CONDITIONS FOR OPERATION

F. Group Scram Insertion Times

Footnote (a) to Table 3.1.4-1

The average of the scram insertion times, from the fully withdrawn position, for the three fastest control rods of all groups of four control rods in a two-by-two array, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

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

M.2

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

ACTION A With the average scram insertion times of control rods exceeding the above limits:

1. Declare the control rods exceeding the above average scram insertion times inoperable until an analysis is performed to determine that required scram reactivity remains for the slow four control rod group, and

2. When operation is continued with an average scram insertion time(s) in excess of the average scram insertion time limit, perform Surveillance Requirement 4.3.D.3 at least once per 60 days of power operation.

With the provisions of the ACTION(s) above not met, be in at least HOT SHUTDOWN within 12 hours.

M.2

4.3 - SURVEILLANCE REQUIREMENTS

F. Group Scram Insertion Times

All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.3.D.

SR 3.1.4.1, SR 3.1.4.2, and SR 3.1.4.4

add proposed LCO 3.1.4 and Table 3.1.4-1

M.2

Page 3 of 3

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 CTS 4.3.D.2 footnote (a), which states that the provisions of Specification 4.0.D are not applicable, has been deleted since proposed SR 3.0.4 provides this allowance (i.e., this allowance has been moved to SR 3.0.4). Therefore, deletion of this allowance is administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 An additional Surveillance Requirement, SR 3.1.4.3, is proposed. This new Surveillance Requirement will require a scram time test, which may be done at any reactor pressure, prior to declaring the control rod operable (and thus, enabling its withdrawal during a startup). To allow testing at less than normal operating pressures, a requirement for scram time limits at < 800 psig is included (ITS Table 3.1.4-1 footnote (b)). These limits appear less restrictive than the operating limits; however, due to reactor pressure not being available to assist the scram speed, the limits are reasonable for application as a test of operability at these conditions. This ensures the affected control rod retains adequate scram performance over the range of applicable reactor pressure. Since this test, and therefore any limits, are not applied in the existing Specification, this is an added restriction. In addition, the reactor pressure applicability of CTS 4.3.D (proposed SRs 3.1.4.1, 3.1.4.2, and 3.1.4.4) has been changed from > 800 psig to ≥ 800 psig for consistency with the new proposed Surveillance.

M.2 The purpose of the control rod scram time LCOs is to ensure the negative scram reactivity corresponding to that used in licensing basis calculations is supported by individual control rod drive scram performance distributions allowed by the Technical Specifications. CTS 3.3.D, 3.3.E, and 3.3.F accomplish the above purpose by placing requirements on maximum individual control rod drive scram times (7 second requirement), average scram times, and local scram times (four control rod group).

TECHNICAL CHANGES - MORE RESTRICTIVE

M.2 (cont'd) Because of the methodology used in the design basis transient analysis (one-dimensional neutronics), all control rods are assumed to scram at the same speed, which is the analytical scram time requirement. Performing an evaluation assuming all control rods scram at the analytical limit results in the generation of a scram reactivity versus time curve, the analytical scram reactivity curve. The purpose of the scram time LCO is to ensure that, under allowed plant conditions, this analytical scram reactivity will be met. Since scram reactivity cannot be readily measured at the plant, the safety analyses use appropriately conservative scram reactivity versus insertion fraction curves to account for the variation in scram reactivity during a cycle. Therefore, the Technical Specifications must only ensure the scram times are satisfied.

The first obvious result is that, if all control rods scram at least as fast as the analytical limit, the analytical scram reactivity curve will be met. However, a distribution of scram times (some slower and some faster than the analytical limit) can also provide adequate scram reactivity. By definition, for a situation where all control rods do not satisfy the analytical scram time limits, the condition is acceptable if the resulting scram reactivity meets or exceeds the analytical scram reactivity curve. This can be evaluated using models which allow for a distribution of scram speeds. It follows that the more control rods that scram slower than the analytical limit, the faster the remaining control rods must scram to compensate for the reduced scram reactivity rate of the slower control rods. ITS 3.1.4 incorporates this philosophy by specifying scram time limits for each individual control rod instead of limits on the average of all control rods and the average of three fastest rods in all four control rod groups. This philosophy has been endorsed by the BWR Owners' Group and described in EAS-46-0487, "Revised Reactivity Control Systems Technical Specifications." The scram time limits listed in ITS Table 3.1.4-1 have margin to the analytical scram time limits listed in EAS-46-0487, Table 3-4 to allow for a specified number and distribution of slow control rods, a single stuck control rod and an assumed single failure. Therefore, if all control rods met the scram time limits found in ITS Table 3.1.4-1, the analytical scram reactivity assumptions are satisfied. If any control rods do not meet the scram time limits, ITS 3.1.4 specifies the number and distribution of these "slow" control rods to ensure the analytical scram reactivity assumptions are still satisfied.

If the number of slow rods is more than 12 or the rods do not meet the separation requirements, the unit must be shutdown within 12 hours. This change is considered more restrictive on plant operation since the proposed individual times are more restrictive than the average times. That is, currently, the "average time" of all rods or a group can be improved by a few fast

TECHNICAL CHANGES - MORE RESTRICTIVE

M.2 (cont'd) scramming rods, even when there may be more than 12 slow rods, as defined in the proposed Specification. Therefore, ITS 3.1.4 limits the number of slow rods to 12 and ensures no more than 2 slow rods occupy adjacent locations.

The maximum scram time requirement in CTS 3.3.D has been retained in ITS 3.1.3 for the purpose of defining the threshold between a slow control rod and an inoperable control rod even though the analyses to determine the LCO scram time limits assumed slow control rods did not scram. Note 2 to ITS Table 3.1.4-1 ensures that a control rod is not inadvertently considered "slow" when the scram time exceeds 7 seconds.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 Proposed SR 3.1.4.2 will test a "representative sample" of control rods each 120 days of power operation instead of the CTS 4.3.D.3 Surveillance Requirement of "10% of the control rods on a rotating basis". The details of what constitutes a representative sample are proposed to be relocated to the Bases. ITS 3.1.4 and SR 3.1.4.2 are adequate to ensure scram time testing is performed. Therefore, the relocated details of what constitutes a representative sample are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

"Specific"

L.1 CTS 4.3.D.1.a requires control rod scram time testing for all control rods prior to exceeding 40% RTP following CORE ALTERATIONS. This effectively means that even if only one bundle is moved (e.g., replacing a leaking fuel bundle mid-cycle), all the control rods are required to be tested. Proposed SR 3.1.4.4 requires control rod scram time testing for only affected control rods following any fuel movement within the affected core cell. This change is acceptable since the intent of testing all of the control rods following CORE ALTERATIONS ensures the overall negative reactivity insertion rate is maintained following refueling activities that may impact a significant number of control rods (e.g., CRD replacement, CRDM overhaul, or movement of fuel in the core cell). When only a few control rods have been impacted by fuel

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 (cont'd) movement, the effect on the overall negative reactivity insertion rate is insignificant. Therefore, it is not necessary to perform scram time testing for all control rods when only a few control rods have been impacted by fuel movement in the reactor pressure vessel. During a routine refueling outage, it is expected that all core cells will be impacted, thus all control rods will be tested, consistent with current requirements. This fact is stated in the Bases for SR 3.1.4.4. The Surveillances of ITS 3.1.4 are adequate to ensure that the negative reactivity insertion rate assumed in the safety analyses is maintained. Additionally, the reliability of the control rods is increased since this change eliminates unnecessary testing for the control rods.

RELOCATED SPECIFICATIONS

None

A.1

ITS 3.1.5

3.3 - LIMITING CONDITIONS FOR OPERATION

G. Control Rod Scram Accumulators

LCO 3.1.5 All control rod scram accumulators shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 5[(a)] A.2 moved to ITS 3.9.5

A.3 add proposed ACTIONS Note

ACTION:

1. In OPERATIONAL MODE 1 or 2:

ACTION A a. With one control rod scram accumulator inoperable, within 8 hours: with reactor steam dome pressure ≥ 900 psig M.1

   1) Restore the inoperable accumulator to OPERABLE status, or A.4

   add proposed Required Action A.1 L.1

   Required Action A.2 2) Declare the control rod associated with the inoperable accumulator inoperable.

b. ~~With the provisions of ACTION 1.a above not met, be in at least HOT SHUTDOWN within the next 12 hours.~~ A.5 M.1

ACTION B ACTION C c. With more than one control rod scram accumulator inoperable, declare the associated control rods inoperable and: add proposed Required Action B.2.1 L.1

Required Actions B.2.2, C.2 within 1 hour L.1

4.3 - SURVEILLANCE REQUIREMENTS

G. Control Rod Scram Accumulators

SR 3.1.5.1 Each control rod scram accumulator shall be determined OPERABLE at least once per 7 days by verifying that the indicated pressure is ≥940 psig ~~unless the control rod is fully inserted and disarmed, or scrammed.~~ A.6

A.2 moved to LCO 3.9.5

a In OPERATIONAL MODE 5, this Specification is applicable for the accumulators associated with each withdrawn control rod and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

3.3 - LIMITING CONDITIONS FOR OPERATION

4.3 - SURVEILLANCE REQUIREMENTS

Required Action D.1 Note

Required Actions B.1 and C.1

ACTION D

1) If the control rod associated with any inoperable scram accumulator is withdrawn, immediately verify that at least one control rod drive pump is operating ~~by inserting at least one withdrawn control rod at least one notch.~~ With no control rod drive pump operating, immediately place the reactor mode switch in the Shutdown position.

L.2

A.7

2) ~~Fully insert the inoperable control rods and disarm the associated directional control valves[b] either:~~

~~a) Electrically, or~~

~~b) Hydraulically by closing the drive water and exhaust water isolation valves.~~

A.8

~~d. With the provisions of ACTION 1.c.2 above not met, be in at least HOT SHUTDOWN within 12 hours.~~

2. In OPERATIONAL MODE 5[a]

a. With one withdrawn control rod with its associated scram accumulator inoperable, fully insert the affected control rod and disarm the associated directional control valves[b] within one hour, either:

A.2 moved to ITS 3.9.5

A.2 moved to ITS 3.9.5

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a In OPERATIONAL MODE 5, this Specification is applicable for the accumulators associated with each withdrawn control rod and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

A.1

ITS 3.1.5

3.3 - LIMITING CONDITIONS FOR OPERATION

4.3 - SURVEILLANCE REQUIREMENTS

1) Electrically, or

2) Hydraulically by closing the drive water and exhaust water isolation valves.

b. With more than one withdrawn control rod with the associated scram accumulator inoperable or no control rod drive pump operating, immediately place the reactor mode switch in the Shutdown position.

A.2 moved to ITS 3.9.5

Page 3 of 3

DISCUSSION OF CHANGES
ITS: 3.1.5 - CONTROL ROD SCRAM ACCUMULATORS

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 The CTS 3.3.G requirements, including Action 2, for control rod scram accumulator OPERABILITY in MODE 5 are being moved to Section 3.9 in accordance with the format of the BWR ISTS, NUREG-1433, Revision 1. Any technical changes to these requirements will be discussed in the Discussion of Changes for ITS: 3.9.5.

A.3 A proposed ACTIONS Note, "Separate Condition entry is allowed for each control rod scram accumulator," has been added to CTS 3.3.G Actions (ITS 3.1.5 ACTIONS) and provides more explicit instructions for proper application of the ACTIONS for Technical Specifications compliance. In conjunction with proposed Specification 1.3, "Completion Times," this Note provides direction consistent with the intent of the existing ACTIONS for inoperable control rod accumulators. Upon discovery of each inoperable accumulator, each specified ACTION is applied, regardless of previous application to other inoperable accumulators.

A.4 The revised presentation of CTS 3.3.G Action 1.a.1) (based on the BWR ISTS, NUREG-1433) does not explicitly detail options to "restore...to OPERABLE status." This action is always an option, and is implied in all Actions. Omitting this action from the ITS is purely editorial.

A.5 ITS 3.1.5 does not contain the equivalent "default" action ("be in at least HOT SHUTDOWN within the next 12 hours") for failure to perform the CTS 3.3.G Action 1.a to declare the associated control rod inoperable. There are no circumstances which preclude the possibility of compliance with an ACTION to "Declare the control rod...inoperable." Therefore, deletion of this "default" action is inconsequential and considered administrative.

A.6 These conditions of CTS 4.3.G, which specify when the accumulator Surveillance does not have to be performed (i.e., when the associated control rod is inserted and disarmed or scrammed), are duplicative of the allowance currently provided by Specification 4.0.C and proposed SR 3.0.1. Therefore, the stated exception has been deleted.

ADMINISTRATIVE (continued)

A.7 The CTS 3.3.G Action 1.c.1) requirement to verify that a control rod drive pump is operating has been maintained, but the method for verifying this has been changed from inserting one control rod one notch to verifying that charging water header pressure is at least 940 psig. These methods both assure that sufficient control rod drive pressure exists to insert the control rods. The proposed method for determining charging water header pressure provides added assurance that the charging water pressure is sufficient to insert all control rods, whereas the existing method only assures that one rod can be inserted. Since the change is merely exchanging one test method for another equivalent (or better) test method, this change is considered administrative.

A.8 CTS 3.3.G Action 1.c requires the affected control rod to be declared inoperable. Once declared inoperable, the CTS 3.3.C Actions for an inoperable control rod are required to be taken. The CTS 3.3.G and ITS 3.1.3 ACTIONS for an inoperable control rod contain requirements to insert and disarm, as well as a shutdown requirement if the Actions are not performed (CTS 3.3.G Action 1.d). The ITS 3.1.5 ACTIONS for inoperable accumulators do not need to repeat the ITS 3.1.3 ACTIONS to insert and disarm, or shutdown the unit if the inoperable control rod is not inserted and disarmed. Therefore CTS 3.3.G Actions 1.c.2 and 1.d have been deleted. Since this change is a presentation preference only, it is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 The ITS 3.1.5 ACTION A for an inoperable control rod accumulator only provides an 8 hour allowance to essentially restore the inoperable accumulator if the reactor pressure is sufficiently high to support control rod insertion. CTS 3.3.G Action 1.a allows 8 hours to restore the inoperable accumulator regardless of the reactor pressure. At reduced reactor pressures, control rods may not insert on a scram signal unless the associated accumulator is OPERABLE. Given the allowances in the proposed LCOs 3.1.3 and 3.1.4 for number and distribution of inoperable and slow control rods, an additional control rod failing to scram (due to inoperable accumulator and low reactor pressure) for up to 8 hours without compensatory action is not justified. Therefore, ITS 3.1.5 ACTION A applies to one inoperable accumulator at sufficiently high reactor pressures. ITS 3.1.5 ACTION C applies to one or more inoperable accumulators at lower reactor pressures. At low reactor pressures, only 1 hour will be provided to restore the inoperable accumulator(s) prior to requiring the associated control rod(s) to be declared inoperable. In addition, charging water header pressure must be $\geq$ 940 psig during this 1 hour, or a reactor scram will be required (ITS 3.1.5 ACTION D).

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

None

"Specific"

L.1 CTS 3.3.G Action 1.a.2) requires a control rod to be declared inoperable within 8 hours when its associated accumulator is inoperable. An inoperable control rod accumulator affects the associated control rod scram time. However, at sufficiently high reactor pressure, the accumulators only provide a portion of the scram force. With this reactor pressure, the control rod will scram even without the associated accumulator, although probably not within the required scram times. Therefore, the option to declare a control rod with an inoperable accumulator "slow" when reactor pressure is sufficient is proposed (ITS 3.1.5 Required Action A.1) in lieu of declaring the control rod inoperable. Since CTS 3.3.G Action 1.a.2) to declare the control rod inoperable allows the control rod to remain withdrawn and not disarmed, ITS 3.1.5 Required Action A.1 to declare the control rod "slow" is essentially equivalent. The proposed limits and allowances for numbers and distribution of inoperable and slow control rods (found in ITS 3.1.3 and ITS 3.1.4, respectively) are appropriately applied to control rods with inoperable accumulators whether declared inoperable or slow. The option for declaring the control rod with an inoperable accumulator "slow" is restricted (by a Note to ITS 3.1.5 Required Actions A.1 and B.2.1) to control rods not previously known to be slow. This restriction limits the flexibility to control rods not otherwise known to have an impaired scram capability.

Additionally, with more than one accumulator inoperable, ITS 3.1.5 ACTIONS B and C provide actions similar to ITS 3.1.5 ACTION A, instead of the CTS 3.3.G Action 1.c requirement to declare the associated control rod inoperable immediately. The requirement to declare the associated control rod inoperable is maintained (ITS 3.1.5 Required Actions B.2.2 and C.2), as well as an option to declare the associated control rod "slow" (ITS 3.1.5 Required Action B.2.1). This added option is only allowed, however, when a sufficiently high reactor pressure exists, since at high reactor pressure there is adequate pressure to scram the rods, even with the accumulator inoperable. The requirement for declaration of control rods as slow, as described in the paragraph above, or inoperable, is limited to 1 hour in ITS 3.1.5 Required Actions B.2.1, B.2.2, and C.2, as opposed to the current immediate declaration of inoperable in CTS 3.3.G Action 1.c. This provides a reasonable time to attempt investigation

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 (cont'd) and restoration of the inoperable accumulator and is sufficiently short such that it does not increase the risk significance of an ATWS event. Furthermore, the 1 hour will only be allowed provided the control rod drive header pressure alone is sufficient to insert control rods if a scram is required (ITS 3.1.5 Required Actions B.1 and C.1).

L.2 CTS 3.3.G Action 1.c.1) for inoperable scram accumulators applies to all reactor pressure situations, whether normal operating pressure or zero pressure. These two extremes represent significant differences in whether or not a control rod with an inoperable accumulator will scram. ITS 3.1.5 acknowledges this difference and presents ACTIONS more appropriate to the actual plant conditions (in one instance, proposing more restrictive ACTIONS - refer to Discussion of Change M.1 above).

CTS 3.3.G Action 1.c.1) is intended to identify the situation where additional scram accumulators (eventually all accumulators) would be expected to become inoperable. Identification of this sort of common cause is significant in ensuring continued plant safety. In the event reactor pressure is too low, where the control rod with an inoperable accumulator may not scram, it is imperative that immediate action be taken if the charging pressure to all accumulators is lost. This requirement is maintained essentially consistent in ITS 3.1.5 Required Action C.1.

However, in the event reactor pressure is sufficiently high (where the control rod will scram even without the associated accumulator), 20 minutes is proposed in ITS 3.1.5 Required Action B.1 to ensure control rod accumulator charging water pressure is adequate to support maintaining the remaining accumulators OPERABLE. This 20 minutes allows an appropriate time to attempt restoration of charging pressure if it should be lost. This proposed action is deemed more appropriate than the CTS 3.3.G Action 1.c.1) requirement to initiate an immediate reactor scram (by placing the reactor mode switch in the shutdown position). The most likely cause of the loss of charging pressure is a trip of the operating CRD pump. Restart of this pump or of the spare CRD pump would restore charging pressure and avoid the plant transient caused by the immediate scram - a scram initiated while withdrawn control rods with inoperable accumulators are known to exist, and the system necessary for manual control rod insertion is not available. Since control rod scram capability remains viable solely from the operating reactor pressure, and the most likely result of the 20 minute allowance of ITS 3.1.5 Required Action B.1 is expected to be restoration of charging pressure (upon which time inoperable control rods could be manually inserted and disarmed, operation returned to normal, and a scram transient avoided), the proposed change is deemed acceptable.

RELOCATED SPECIFICATIONS

None

ITS 3.1.6

M.1

Insert New Specification 3.1.6

Insert new Specification 3.1.6, "Rod Pattern Control," as shown in the Quad Cities 1 and 2 Improved Technical Specifications.

ADMINISTRATIVE

None

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 A new Specification requiring the control rod pattern to be in compliance with the analyzed rod position sequence when THERMAL POWER is $\leq$ 10% RTP in MODES 1 and 2 is being added. Appropriate ACTIONS and Surveillance Requirements are also added, consistent with the BWR ISTS, NUREG-1433, Revision 1. This change represents an additional restriction on plant operation necessary to ensure the analysis assumptions relative to the Control Rod Drop Accident are maintained.

TECHNICAL CHANGES - LESS RESTRICTIVE

None

RELOCATED SPECIFICATIONS

None

3.4 - LIMITING CONDITIONS FOR OPERATION

LCO 3.1.7 A. Standby Liquid Control System (SLCS)

The standby liquid control system (SLCS) shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

ACTION A 1. With one subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least

ACTION C HOT SHUTDOWN within the next 12 hours.

ACTION B 2. With both standby liquid control subsystems inoperable, restore at least one subsystem to OPERABLE status within 8 hours ~~or be in at least HOT~~

ACTION C SHUTDOWN within the next 12 hours.

4.4 - SURVEILLANCE REQUIREMENTS

A. Standby Liquid Control System

The standby liquid control system shall be demonstrated OPERABLE:

1. At least once per 24 hours by verifying that:

SR 3.1.7.2 a. The temperature of the sodium pentaborate solution is greater than or equal to the limits of Figure 3.4.A-1.

SR 3.1.7.1 b. The volume of the sodium pentaborate solution is greater than or equal to the limits shown in Figure 3.4.A-2.

SR 3.1.7.3 c. The temperature of the pump suction piping to be greater than or equal to 83°F.

2. At least once per 31 days by:

SR 3.1.7.4 a. Verifying the continuity of the explosive charge.

SR 3.1.7.5 b. Determining[a] ~~by chemical analysis~~ that the available concentration of boron in solution is ~~14% by weight to 16.5% by weight.~~ within limits of Figure 3.1.7-1

LA.1

A.3

SR 3.1.7.6 c. Verifying that each valve, manual, ~~power operated or automatic,~~ 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.

A.2

once within 24 hours after M.1

SR 3.1.7.5 a This surveillance shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below the limits specified by Figure 3.4.A-1.

3.4 - LIMITING CONDITIONS FOR OPERATION

4.4 - SURVEILLANCE REQUIREMENTS

SR 3.1.7.7

3. When tested pursuant to Specification 4.0.E, by demonstrating that the minimum flow requirement of 40 gpm per pump at a pressure of greater than or equal to 1275 psig is met.

4. At least once per ~~18~~ 24 months by: LD.1

SR 3.1.7.8

a. Initiating one of the standby liquid control subsystems, ~~including an explosive valve,~~ LA2 and verifying that a flow path from the pumps to the reactor pressure vessel is available. Both injection loops shall be tested in ~~36~~ 48 months. LD.1

~~b. Deleted~~

SR 3.1.7.9

c. Demonstrating that the pump suction line from the storage tank is not plugged.

add second Frequency

M.2

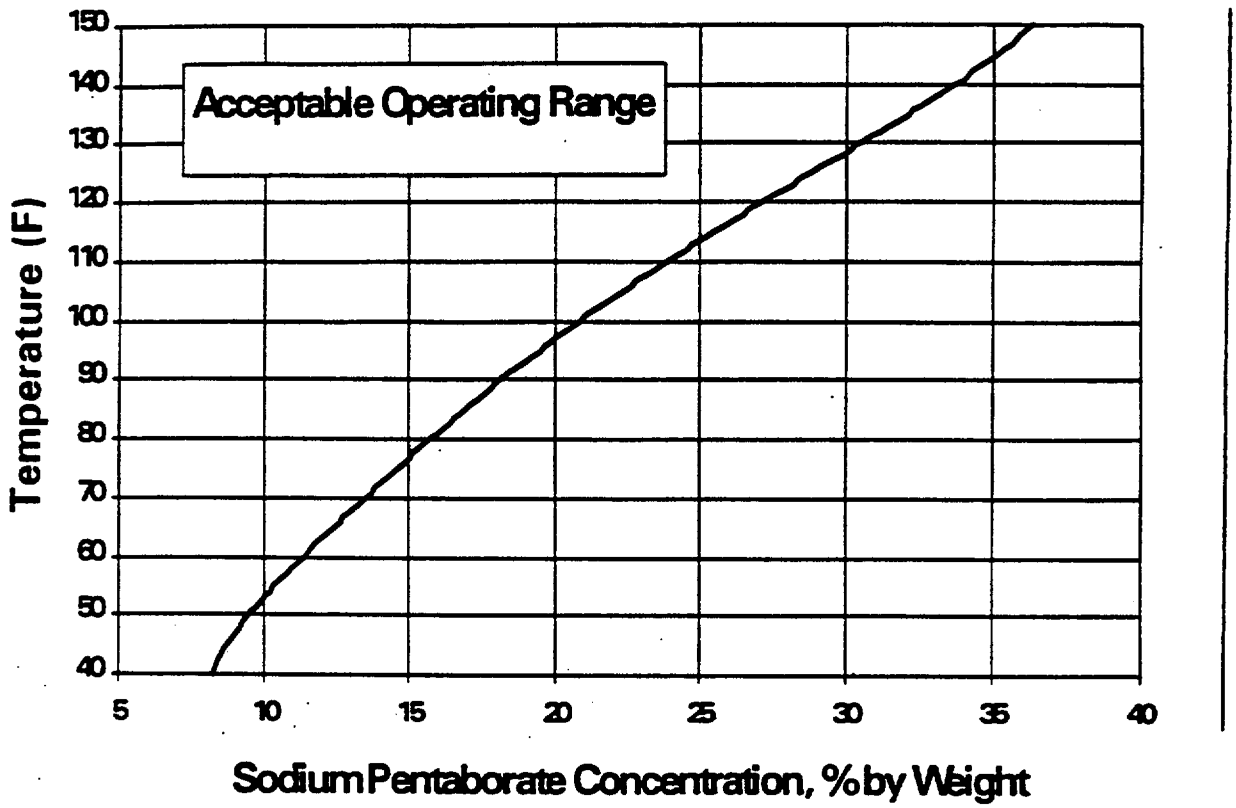
A.1

ITS 3.1.7

Figure 3.1.7-2

FIGURE 3.4.A-1

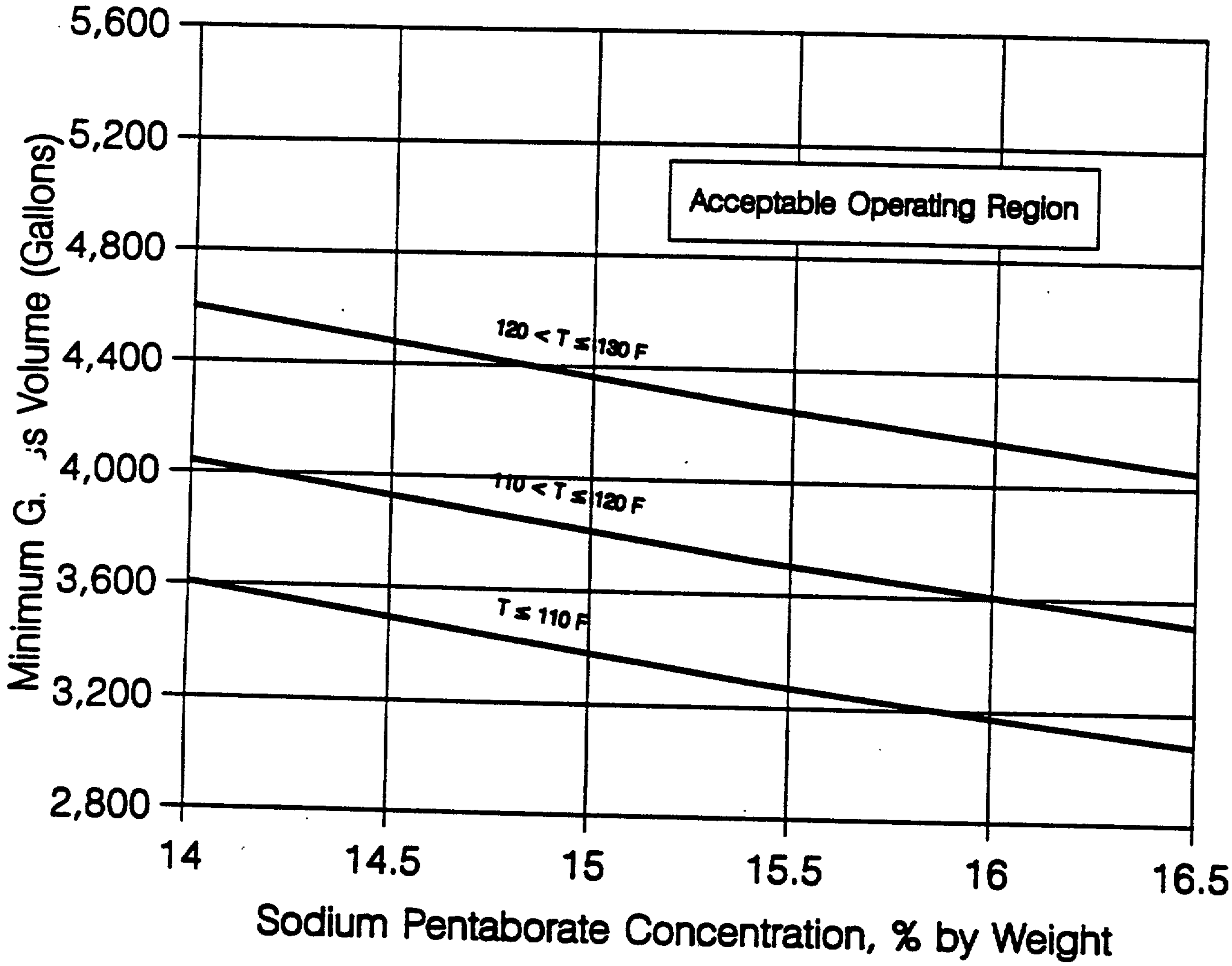
SODIUM PENTABORATE SOLUTION TEMPERATURE REQUIREMENTS



Page 3 of 4

FIGURE 3.4.A-2

SODIUM PENTABORATE SOLUTION VOLUME REQUIREMENTS



ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 CTS 4.4.A.2.c requires the verification every 31 days that each manual, power operated, or 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. Since the only "power operated or automatic" valves in the system is the explosive valve, the requirement to verify the "power operated or automatic" valves is deleted. The continuity of the explosive charge is verified in CTS 4.4.A.2.a (proposed SR 3.1.7.4). Since there are no differences in the performance of the actual Surveillance, this change is considered administrative in nature.

A.3 The details of CTS 4.4.A.2.b, which identify the available boron concentration to be determined to be 14% to 16.5% by weight, are revised in proposed SR 3.1.7.5 to be within the limits of Figure 3.1.7-1. Since the limits identified in the Figure correspond to the same 14% to 16.5% by weight related to the volume requirement, this change is considered a presentation preference consistent with the BWR ISTS, NUREG-1433, Rev. 1. Therefore, this change is administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS 4.4.A.2.b requires the determination that the available concentration of sodium pentaborate in solution is within limits every 31 days and in accordance with footnote a (anytime water or boron is added to the solution or when the system temperature drops below the limits). This Surveillance is retained in proposed SR 3.1.7.5; however, a requirement has been added to require the Surveillance in footnote a to be completed within 24 hours. This ensures that any potential change to the boron concentration is quickly evaluated. Since an explicit time limitation is provided this change is considered more restrictive.

M.2 CTS 4.4.A.4.c requires the demonstration that the pump suction line from the storage tank is not plugged. This Surveillance is retained in proposed SR 3.1.7.9. A new requirement has been added to perform this Surveillance once within 24 hours after piping temperature is restored within the limits of ITS

TECHNICAL CHANGES - MORE RESTRICTIVE

M.2 (cont'd) Figure 3.1.7-2 (CTS Figure 3.4.A-1). This change is considered more restrictive since an explicit Surveillance will be required whenever the limits of Figure 3.1.7-2 are not met. However, this change is necessary since precipitation of the boron from solution may occur when the temperature requirements are not met.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The detail of the method for performing CTS 4.4.A.2.b, the Surveillance to determine boron concentration is within limits (by a chemical analysis), is proposed to be relocated to the Bases. This detail is not necessary to ensure that SLC System is maintained OPERABLE. The requirements of ITS 3.1.7 and SR 3.1.7.5 are adequate to ensure the boron concentration is within limits and to ensure SLC System OPERABILITY. Therefore, the relocated detail is not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

LA.2 The detail of the method for performing CTS 4.4.A.4.a, the Surveillance to verify flow through the SLC subsystem into the reactor pressure vessel (initiating an explosive valve), is proposed to be relocated to the Bases. This detail is not necessary to ensure the SLC System is maintained OPERABLE. The requirements of ITS 3.1.7 and SR 3.1.7.8 are adequate to ensure the capability to provide flow through each SLC subsystem into the reactor pressure vessel and to ensure SLC System OPERABILITY. Therefore, the relocated detail is not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS .

LD.1 The Frequency for performing CTS 4.4.A.4.a and 4.4.A.4.c (proposed SRs 3.1.7.8 and 3.1.7.9) has been extended from 18 months to 24 months. These SRs ensure that the SLC System is capable of injecting into the reactor pressure vessel by verifying a flow path and also by firing one of the explosive valves. The proposed change will allow these Surveillances to extend their Surveillance Frequency from the current 18 month Surveillance Frequency (36 months for CTS 4.4.A.4.a) (i.e., a maximum of 22.5 months (45 months for CTS 4.4.A.4.a) accounting for the allowable grace period specified in CTS 4.0.B and proposed SR 3.0.2) to a 24 month Surveillance Frequency

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (48 months for SR 3.1.7.8) (i.e., a maximum of 30 months (60 months for SR 3.1.7.8) accounting for the allowable grace period specified in CTS 4.0.B and proposed SR 3.0.2). This proposed change was evaluated in accordance with the guidance provided in NRC Generic Letter No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Reviews of historical maintenance and surveillance data have shown that these tests normally pass their Surveillances at the current Frequency. An evaluation has been performed using this data, and it has been determined that the effect on safety due to the extended Surveillance Frequency will be minimal. This conclusion is based on the following evaluation. As described in the ITS Bases, the SLC System is a backup safety system to the Control Rod Drive (CRD) System. In the event of a low probability failure of the CRD System, the SLC System is designed to bring the reactor subcritical during the most reactive point in core life. The SLC System is designed so that all active components are single failure proof. In addition, each of the SLC System pumps is tested during the operating cycle in accordance with SR 3.1.7.7 (Inservice Testing Program) which verifies system capacity. SR 3.1.7.2 and SR 3.1.7.3 ensure the temperature in the SLC system tank and SLC pump suction piping is maintained to prevent the precipitation of sodium pentaborate. SR 3.1.7.4 verifies the continuity of the charge in the explosive valves. These tests ensure that the SLC System is Operable during the operating cycle. Finally, the explosive valves are designed to be highly reliable. Based on the inherent system and component reliability, and the testing performed during the operating cycle, the impact, if any, from this change on system availability is minimal. The review of historical surveillance data also demonstrated that there are no failures that would invalidate the conclusion that the impact, if any, on system availability is minimal from a change to CTS 4.4.A.4.a and 4.4.A.4.c as implemented in SR 3.1.7.8 and SR 3.1.7.9. In addition, the proposed 24 month Surveillance Frequencies (48 months for SR 3.1.7.8), if performed at the maximum interval allowed by proposed SR 3.0.2 (30 months or 60 months, as applicable) do not invalidate any assumptions in the plant licensing basis.

"Specific"

None

RELOCATED SPECIFICATIONS

None

3.3 - LIMITING CONDITIONS FOR OPERATION

K. SDV Vent and Drain Valves

LCO 3.1.8 All scram discharge volume (SDV) vent and drain valves shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

ACTION A 1. With[b] one or more SDV vent or drain lines with one valve inoperable, isolate[c] the associated line within 7 days ~~or be in HOT SHUTDOWN~~ within the next 12 hours. ACTION C

ACTION B 2. With[b] one or more SDV vent or drain lines with both valves inoperable, isolate[c] the associated line within 8 hours or be in HOT SHUTDOWN within the next 12 hours. ACTION C

4.3 - SURVEILLANCE REQUIREMENTS

K. SDV Vent and Drain Valves

The scram discharge volume vent and drain valves shall be demonstrated OPERABLE:

SR 3.1.8.1 1. At least once per 31 days by verifying each valve to be open[a], and

SR 3.1.8.2 2. At least once per 92 days by cycling each valve through at least one complete cycle of travel.

SR 3.1.8.3 3. At least once per 18 24 months, the scram discharge volume vent and drain valves shall be demonstrated to: LA.1

a. Close within 30 seconds after receipt of a actual or simulated signal for control rods to scram, and

b. Open after the actual or simulated scram signal is reset.

A.2

ACTIONS Note 1 — b Separate Action statement entry is allowed for each SDV vent and drain line.

ACTIONS Note 2 — c An isolated line may be unisolated under administrative control to allow draining and venting of the SDV.

Note to SR 3.1.8.1 — a These valves may be closed intermittently for testing under administrative controls.

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 The phrase "actual or simulated" in reference to the signal used for performing CTS 4.3.K.3.a and CTS 4.3.K.3.b (proposed SR 3.1.8.3), is proposed to be added. OPERABILITY is adequately demonstrated in either case since the SDV vent and drain valves cannot discriminate between "actual" or "simulated" signals. This change only clarifies the type of signal that may be used to perform the Surveillance Requirement and is therefore considered to be administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LD.1 The Frequency for performing CTS 4.3.K.3 (proposed SR 3.1.8.3) has been extended from 18 months to 24 months. This SR ensures that the vent and drain valves close in ≤ 30 seconds after receipt of an actual or simulated scram signal; and open when the actual or simulated scram signal is reset. The proposed change will allow this Surveillance to extend its Surveillance Frequency from the current 18 month Surveillance Frequency (i.e., a maximum of 22.5 months accounting for the allowable grace period specified in CTS 4.0.B and proposed SR 3.0.2) to a 24-month Surveillance Frequency (i.e., a maximum of 30 months accounting for the allowable grace period specified in CTS 4.0.B and proposed SR 3.0.2). This proposed change was evaluated in accordance with the guidance provided in NRC Generic Letter No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Reviews of historical maintenance and surveillance data have shown that these tests normally pass their Surveillances at the current Frequency. An evaluation has been performed using this data, and it has been determined that the effect on safety due to the extended Surveillance Frequency will be minimal.

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd)

SR 3.1.8.2 requires that the SDV vent and drain valves be cycled fully closed and fully open every 92 days during the operating cycle. SR 3.1.8.2 ensures that the mechanical components and a portion of the valve logic remains operable. This test does not ensure that the logic of the SDV vent and drain valves is operable, but logic systems are inherently more reliable. This is acknowledged in the NRC safety evaluation report, dated August 2, 1993, relating to the extension of Peach Bottom Atomic Power Station, Units number 2 and 3, surveillance interval extension from 18 to 24 months.

> "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC 30936P) show that the overall reliability of safety systems' reliabilities are not dominated by the reliabilities of the logic systems, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical component failure, increasing the logic system functional test interval represents no significant change in the overall safety system unavailability."

Because of the inherent equipment reliability (as demonstrated by years of operating experience in the nuclear and non-nuclear industry), more frequent stroke testing of the subject valves, it is concluded that the impact, if any, on system availability is minimal as a result of this change.

The review of historical surveillance data also demonstrated that there are no failures that would invalidate the conclusion that the impact, if any, on system availability is minimal from a change to a 24 month operating cycle. In addition, performing the SR at the maximum interval allowed by proposed SR 3.0.2 does not invalidate any assumptions in the plant licensing basis.

"Specific"

None

RELOCATED SPECIFICATIONS

None

3.3 - LIMITING CONDITIONS FOR OPERATION

J. Control Rod Drive Housing Support

The control rod drive housing support shall be in place.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 3.

ACTION:

With the control rod drive housing support not in place, be in at least HOT SHUTDOWN within 12 hours and in at least COLD SHUTDOWN within the following 24 hours.

4.3 - SURVEILLANCE REQUIREMENTS

J. Control Rod Drive Housing Support

The control rod drive housing support shall be verified to be in place by a visual inspection prior to startup any time it has been disassembled or when maintenance has been performed in the control rod drive housing support area.

L.1

DISCUSSION OF CHANGES
CTS: 3/4.3.J - CONTROL ROD DRIVE HOUSING SUPPORT

ADMINISTRATIVE

None

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

None

"Specific"

L.1 The CTS 3/4.3.J requirement for the Control Rod Drive Housing Support to be in place is included in the OPERABILITY requirements for control rods. Plant configuration management provides adequate controls to assure the CRD housing support is in place. The current Technical Specifications require inspections of the CRD housing support prior to startup following reassembly. This current Technical Specifications requirement verifies that the CRD housing support is in place for reactor operation in MODES 1, 2, and 3. Post-maintenance inspections conducted through plant configuration management control have the same function as the current Technical Specifications requirement. Since work is not normally performed on the CRD housing support at power, and checks on its installation are not made at power there is no current requirement to verify CRD housing support installation in power operating conditions. Therefore, the deletion of this current Technical Specifications is acceptable based on use of plant configuration management control to ensure proper CRD housing support installation.

RELOCATED SPECIFICATIONS

None

3.3 - LIMITING CONDITIONS FOR OPERATION

N. Economic Generation Control (EGC) System

The economic generation control (EGC) system may be in operation with automatic flow control provided:

1. Core flow is within 65% to 100% of rated core flow, and
2. THERMAL POWER is ≥20% of RATED THERMAL POWER.

APPLICABILITY

OPERATIONAL MODE 1.

ACTION:

With core flow less than 65% or greater than 100% of rated core flow, or THERMAL POWER less than 20% of RATED THERMAL POWER, restore operation to within the limits within one hour. Otherwise, immediately remove the plant from EGC operation.

4.3 - SURVEILLANCE REQUIREMENTS

N. Economic Generation Control (EGC) System

The economic generation control system shall be demonstrated OPERABLE by verifying that core flow is within 65% to 100% of rated core flow and THERMAL POWER is ≥20% of RATED THERMAL POWER:

1. Prior to entry into EGC operation, and
2. At least once per 12 hours while operating in EGC.

ADMINISTRATIVE

None

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

None

RELOCATED SPECIFICATIONS

R.1 The Economic Generation Control System was designed to allow the load dispatcher to control power output of the station within constraints of the system design. These constraints are well within the analyzed system setpoints utilized in DBA and transient analyses. The Economic Generation Control System is not assumed in any of these analyses. Therefore, the requirements specified in CTS 3/4.3.N did not satisfy the NRC Final Policy Statement Technical Specification screening criteria as documented in the Application Selection Criteria to the Quad Cities 1 and 2 Technical Specifications, and have been relocated to the Technical Requirements Manual (TRM). The TRM will be incorporated by reference into the UFSAR at ITS implementation. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59.

DISCUSSION OF CHANGES
ITS: SECTION 3.1 - REACTIVITY CONTROL SYSTEM BASES

The Bases of the current Technical Specifications for this section (B 3/4.3-1 through B 3/4.3-7 and B 3/4.4-1 through B 3/4.4-2) have been completely replaced by revised Bases reflecting the format and applicable content of the Quad Cities 1 and 2 ITS Section 3.1, consistent with the BWR ISTS, NUREG-1433, Rev. 1. The revised Bases are as shown in the Quad Cities 1 and 2 ITS Bases.

<CTS>

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 SDM shall be:

<3.3.A>

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

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

1

<Appl 3.3.A>

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

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.3.A Act 1>	A. SDM not within limits in MODE 1 or 2.	A.1 Restore SDM to within limits.	6 hours
<3.3.A Act 1>	B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	12 hours
<3.3.A Act 2>	C. SDM not within limits in MODE 3.	C.1 Initiate action to fully insert all insertable control rods.	Immediately
<3.3.A Act 2>	D. SDM not within limits in MODE 4.	D.1 Initiate action to fully insert all insertable control rods. AND	Immediately

(continued)

<CTS>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. (continued) <3.3.A Act 2>	D.2 Initiate action to restore [secondary] containment to OPERABLE status.	1 hour [2]
	AND	
	D.3 Initiate action to restore one standby gas treatment (SGT) subsystem 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 [2]
E. SDM not within limits in MODE 5. <3.3.A Act 3>	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)

⟨CTS⟩

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. (continued) ⟨3.3.A Act 3⟩	E.3	Initiate action to restore [secondary] containment to OPERABLE status.	1 hour 2
	AND		
	E.4	Initiate action to restore one SGT subsystem to OPERABLE status.	1 hour
	AND		
	E.5	Initiate action to restore isolation capability in each required [secondary] containment penetration flow path not isolated.	1 hour 2

<CTS>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify SDM is: <4.3.A> a. ≥ [0.38]% Δk/k with the highest worth control rod analytically determined; or b. ≥ [0.28]% Δk/k with the highest worth control rod determined by test. 1	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

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.1.1 - SHUTDOWN MARGIN

1. TSTF-9 relocates SHUTDOWN MARGIN (SDM) limits of NUREG-1433, Revision 1 Specification 3.1.1 to the CORE OPERATING LIMITS REPORT (COLR). The justification for this change states that SDM is a cycle specific variable. At Quad Cities 1 and 2 SDM limits are not cycle specific. Therefore, the TSTF-9 is not incorporated into ITS 3.1.1 and the SDM limits are maintained in the Technical Specifications. The brackets for the limits have been removed and the proper plant specific value has been provided.

2. The brackets have been removed and the proper plant specific information/value has been provided.

<CTS>

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Reactivity Anomalies

<3.3.B>

LCO 3.1.2 The reactivity {difference} between the {monitored ~~rod density~~ core keff and the predicted ~~rod density~~} core keff shall be within ± 1% Δk/k. 1

<Appl 3.3.B>

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.3.B Act> A. Core reactivity {difference} not within limit. 1	A.1 Restore core reactivity {difference} to within limit. 1	72 hours
<3.3.B Act> B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

<CTS>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	Verify core reactivity [difference] between the [monitored rod density and the predicted rod density] is within ± 1% Δk/k. 1 core keff 1	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 operations in MODE 1

<4.3.B>

1. The brackets have been removed and the proper plant specific information/value has been provided.

⟨CTS⟩

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Control Rod OPERABILITY

⟨3.3.C⟩ LCO 3.1.3 Each control rod shall be OPERABLE.

⟨Appl 3.3.C⟩ APPLICABILITY: MODES 1 and 2.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each control rod.

⟨3.3.C Act 1⟩ ⟨3.3.C Act 1.a.2⟩ ⟨3.3.D Act⟩

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One withdrawn control rod stuck.	------------NOTE------------ Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation," if required, to allow continued operation. ----------------------------	
	A.1 Verify stuck control rod separation criteria are met. AND	Immediately
	A.~~1~~2 Disarm the associated control rod drive (CRD). TSTF-32 AND	2 hours
		(continued)

<CTS>

TSTF-32

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued) <4.3.C.1.b> TSTF-33 from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM <3.3.C Act 1.c> <4.3.A.2>	A.~~2~~ 3 ~~--------NOTE--------- Not applicable when less than or equal to the low power setpoint (LPSP) of the RWM. ---------------------~~ Perform SR 3.1.3.2 and SR 3.1.3.3 for each withdrawn OPERABLE control rod. AND A.~~3~~ 4 Perform SR 3.1.1.1. TSTF-32	TSTF-33 24 hours 72 hours
B. Two or more withdrawn control rods stuck. <DOC M.3>	~~B.1 Disarm the associated CRD.~~ ~~AND~~ B.~~2~~ 1 Be in MODE 3.	~~2 hours~~ TSTF-34 12 hours
C. One or more control rods inoperable for reasons other than Condition A or B. <3.3.C Act 2> <3.3.C Act 2.a.2> <3.3.D Act> <3.3.H Act 1.b> <3.3.I Act 1>	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. AND	 3 hours

(continued)

⟨CTS⟩

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
⟨3.3.C Act 2.b & 2.c⟩ ⟨3.3.H Act 1.b⟩ C. (continued)	C.2 Disarm the associated CRD.	4 hours
⟨3.3.C Act 1.a⟩ ⟨3.3.C Act 2.a.1⟩ D. ---------NOTE--------- Not applicable when THERMAL POWER > ~~[~~10~~]~~% RTP. [1] --------------------- Two or more inoperable control rods not in compliance with ~~banked position withdrawal sequence (BPWS)~~ analyzed rod position sequence [2] and not separated by two or more OPERABLE control rods.	D.1 Restore compliance with ~~BPWS~~. **OR** D.2 Restore control rod to OPERABLE status.	4 hours 4 hours
~~[E. ---------NOTE--------- Not applicable when THERMAL POWER > [10]% RTP. --------------------- One or more groups with four or more inoperable control rods.~~ [1] [4]	~~E.1 Restore control rod to OPERABLE status.~~	~~4 hours]~~
⟨3.3.C Act 1.b⟩ ⟨3.3.C Act 1.c⟩ ⟨3.3.C Act 3⟩ ⟨3.3.C Act 4⟩ ⟨3.3.H Act 2⟩ ⟨3.3.I Act 2⟩ ~~F~~ E [4]. Required Action and associated Completion Time of Condition A, C, D, or ~~E~~ not met. [4] **OR** Nine or more control rods inoperable.	~~F~~ E.1 [4] Be in MODE 3.	12 hours

<CTS>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
<3.3.I> <4.3.I> SR 3.1.3.1	Determine the position of each control rod.	24 hours
SR 3.1.3.2 <4.3.C.1>	-------------------NOTE------------------- Not required to be performed until 7 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of RWM. -- Insert each fully withdrawn control rod at least one notch.	7 days
SR 3.1.3.3 <4.3C.1>	-------------------NOTE------------------- Not required to be performed until 31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM. -- Insert each partially withdrawn control rod at least one notch.	31 days
SR 3.1.3.4 <3.3.D>	Verify each control rod scram time from fully withdrawn to ~~notch/position [06]~~ 90% insertion is ≤ 7 seconds. 3	In accordance with SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4

(continued)

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

<3.3.H>
<4.3.H>

SURVEILLANCE		FREQUENCY
SR 3.1.3.5	Verify each control rod does not go to the withdrawn overtravel position.	Each time the control rod is withdrawn to "full out" position AND Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect coupling

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

3. Quad Cities 1 and 2 evaluate scram time performance based on percent control rod insertion instead of notch position. The percent insertion criterion is being retained consistent with the current licensing basis in order to allow correlation to existing historical scram time data.

4. ISTS 3.1.3 ACTION E is applicable to plants with Siemens Power Corporation (SPC) fuel. Although Quad Cities 1 and 2 use SPC fuel, ComEd performs cycle-specific control rod drop accident (CRDA) analyses that incorporate eight rods out of service with at least two cell separation in order to confirm that energy deposition is less than 280 calories per gram. Consequently, this ACTION is not applicable to Quad Cities 1 and 2 and has been deleted. As a result of this deletion, the following Conditions, Required Actions, and references to the Conditions have been renumbered.

⟨CTS⟩

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Control Rod Scram Times

12 [1]

LCO 3.1.4 ⟨3.3.E⟩⟨M.2⟩ ⟨3.3.F⟩⟨M.2⟩

a. No more than [10] 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.

⟨App 3.3.E⟩ ⟨App 3.3.F⟩ APPLICABILITY: MODES 1 and 2.

ACTIONS

⟨3.3.E Act⟩ ⟨3.3.F Act⟩

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
⟨4.3.D⟩ ⟨4.3.E⟩ 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. [1]	Prior to exceeding 40% RTP after fuel movement within the ~~reactor pressure vessel~~ affected core cell AND TSTF-222
		(continued)

(move to SR 3.1.4.4

⟨CTS⟩

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
⟨4.3.D⟩ ⟨4.3.E⟩	SR 3.1.4.1 (continued)	Prior to exceeding 40% RTP after each reactor shutdown ≥ 120 days
⟨4.3.D⟩ ⟨4.3.E⟩	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. 1	120 days cumulative operation in MODE 1
⟨DOC M.3⟩	SR 3.1.4.3 Verify 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
⟨4.3.D⟩ ⟨4.3.E⟩	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. 1	Insert Frequency from SR 3.1.4.1 TSTF-222 AND TSTF-222 Prior to exceeding 40% RTP after work on control rod or CRD System that could affect scram time

<CTS>

<3.3.E>
<3.3.F>
<DOC M.2>

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]~~ 90% insertion. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

2

PERCENT INSERTION ~~NOTCH POSITION~~	SCRAM TIMES(a)(b) (seconds) when REACTOR STEAM DOME PRESSURE ≥ ~~[800]~~ psig
5 ~~[46]~~	0.36 ~~[0.44]~~
20 ~~[36]~~	0.84 ~~[1.08]~~
50 ~~[26]~~	1.86 ~~[1.83]~~
90 ~~[06]~~	3.25 ~~[3.35]~~

2 1 1 3

(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.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.1.4 - CONTROL ROD SCRAM TIMES

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. Quad Cities 1 and 2 evaluate scram time performance based on percent control rod insertion instead of notch position. The percent insertion criterion is being retained consistent with the current licensing basis in order to allow correlation to existing historical scram time data. The proposed scram times were established consistent with the methodology described in BWROG-8754, "BWR Owner's Group Revised Reactivity Control System Technical Specifications."

3. Editorial change made for enhanced clarity.

<CTS>

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Control Rod Scram Accumulators

<3.3.6>

LCO 3.1.5 Each control rod scram accumulator shall be OPERABLE.

<App 3.3.6>

APPLICABILITY: MODES 1 and 2.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each control rod scram accumulator.

<3.3.6 Action 1>

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One control rod scram accumulator inoperable with reactor steam dome pressure ≥ [900] psig. $\boxed{1}$	A.1	--------NOTE-------- Only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time Surveillance. -------------------- Declare the associated control rod scram time "slow."	8 hours
	<u>OR</u>		
	A.2	Declare the associated control rod inoperable.	8 hours

(continued)

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two or more control rod scram accumulators inoperable with reactor steam dome pressure ≥ ~~[~~900~~]~~ psig. [1]	B.1 Restore charging water header pressure to ≥ ~~[~~940~~]~~ psig. [1]	20 minutes from discovery of Condition B concurrent with charging water header pressure < ~~[~~940~~]~~ psig [1]
	AND	
	B.2.1 --------NOTE--------- Only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time Surveillance. ------------------- Declare the associated control rod scram time "slow."	1 hour
	OR	
	B.2.2 Declare the associated control rod inoperable.	1 hour

<3.3.6 Action 1.c>

(continued)

⟨CTS⟩

ACTIONS (continued)

	CONDITION	REQUIRED ACTION		COMPLETION TIME
⟨3.3.6 Action 1.c⟩	C. One or more control rod scram accumulators inoperable with reactor steam dome pressure < [900] psig. 1	C.1	Verify all control rods associated with inoperable accumulators are fully inserted.	Immediately upon discovery of charging water header pressure < [940] psig 1
		AND		
		C.2	Declare the associated control rod inoperable.	1 hour
⟨3.3.6 Act 1.c.1⟩	D. Required Action and associated Completion Time of Required Action B.1 or C.1 not met. 2	D.1	--------NOTE--------- Not applicable if all inoperable control rod scram accumulators are associated with fully inserted control rods. ---------------------- Place the reactor mode switch in the shutdown position.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
⟨4.3.G⟩	SR 3.1.5.1	Verify each control rod scram accumulator pressure is ≥ [940] psig. 1	7 days

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.1.5 - CONTROL ROD SCRAM ACCUMULATORS

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. Editorial change for clarity or for consistency with the Writer's Guide.

<CTS>

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Rod Pattern Control

analyzed rod position
1

<DOC M.6> LCO 3.1.6 OPERABLE control rods shall comply with the requirements of the ~~[banked position withdrawal]~~ sequence ~~(BPWS)]~~.

APPLICABILITY: MODES 1 and 2 with THERMAL POWER ≤ ~~[~~10~~]~~% RTP.

1

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
<DOC M.6> A. One or more OPERABLE control rods not in compliance with ~~[BPWS]~~. the analyzed rod position sequence 1	A.1	---------NOTE--------- Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation." ---------------------	
		Move associated control rod(s) to correct position.	8 hours
	OR		
	A.2	Declare associated control rod(s) inoperable.	8 hours

(continued)

⟨CTS⟩

ACTIONS (continued)

	CONDITION	REQUIRED ACTION		COMPLETION TIME
⟨DOC M.6⟩ NEDO-21231 page 7-1	B. Nine or more OPERABLE control rods not in compliance with ~~[BPWS]~~ the analyzed rod position sequence $\boxed{1}$.	B.1	--------NOTE-------- Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1. --------------------- Suspend withdrawal of control rods.	Immediately
		AND		
		B.2	Place the reactor mode switch in the shutdown position.	1 hour

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
⟨DOC M.6⟩	SR 3.1.6.1	Verify all OPERABLE control rods comply with ~~[BPWS]~~ the analyzed rod position sequence $\boxed{1}$.	24 hours

1. The brackets have been removed and the proper plant specific information/value has been provided.

⟨CTS⟩

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

⟨3.4.A⟩ LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

⟨Appl 3.4.A⟩ 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~~. One SLC subsystem inoperable ~~[for reasons other than Condition A]~~.	A ~~B~~.1 Restore SLC subsystem to OPERABLE status.	7 days ~~AND 10 days from discovery of failure to meet the LCO~~
B ~~C~~. Two SLC subsystems inoperable ~~[for reasons other than Condition A]~~.	B ~~C~~.1 Restore one SLC subsystem to OPERABLE status.	8 hours
C ~~D~~. Required Action and associated Completion Time not met.	C ~~D~~.1 Be in MODE 3.	12 hours

⟨3.4.A Act 1⟩ ⟨3.4.A Act 2⟩ ⟨3.4.A Act 1⟩ ⟨3.4.A Act 2⟩ 1 1 1 1 1

⟨CTS⟩

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
⟨4.4.A.1.b⟩	SR 3.1.7.1	Verify available volume of sodium pentaborate solution is [within the limits of Figure 3.1.7-1, ~~or ≥ [4530] gallons~~].	24 hours } 2
⟨4.4.A.1.a⟩	[SR 3.1.7.2	Verify temperature of sodium pentaborate solution is within the limits of [Figure 3.1.7-2]. 2	24 hours] 2
⟨4.4.A.1.c⟩	[SR 3.1.7.3	Verify temperature of pump suction piping is ~~within the limits of [Figure 3.1.7-2]~~.	24 hours 2 ≥ 83°F] 2
⟨4.4.A.2.a⟩	SR 3.1.7.4	Verify continuity of explosive charge.	31 days
⟨4.4.A.2.b⟩	SR 3.1.7.5	Verify the concentration of ~~boron~~ sodium pentaborate in solution is [within the limits of Figure 3.1.7-1]. 2 3	31 days **AND** Once within 24 hours after water or boron is added to solution **AND** Once within 24 hours after solution temperature is restored within the limits of [Figure 3.1.7-2] 2

(continued)

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.1.7.6 <4.4.A.2.c>	Verify each SLC subsystem manual, ~~power operated, [and automatic]~~ valve~~s~~ 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. 1	31 days
SR 3.1.7.7 <4.4.A.3>	Verify each pump develops a flow rate ≥ ~~[41.2]~~ 40 gpm at a discharge pressure ≥ ~~[1190]~~ 1275 psig. 2 2	~~[~~In accordance with the Inservice Testing Program ~~or 92 days]~~ 2
SR 3.1.7.8 <4.4.A.4.a>	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	~~[18]~~ 24 months on a STAGGERED TEST BASIS 2
~~[~~SR 3.1.7.9 <4.4.A4.c>	Verify all heat traced piping between storage tank and pump suction is unblocked.	~~[18]~~ 24 months AND Once within 24 hours after ~~solution~~ piping 4 temperature is restored within the limits of ~~[~~Figure 3.1.7-2~~]~~ 2 ~~]~~ 2 2
~~SR 3.1.7.10~~	~~Verify sodium pentaborate enrichment is ≥ [60.0] atom percent B-10.~~ 1	~~Prior to addition to SLC tank~~

Insert Figure 3.1.7-1

[2]

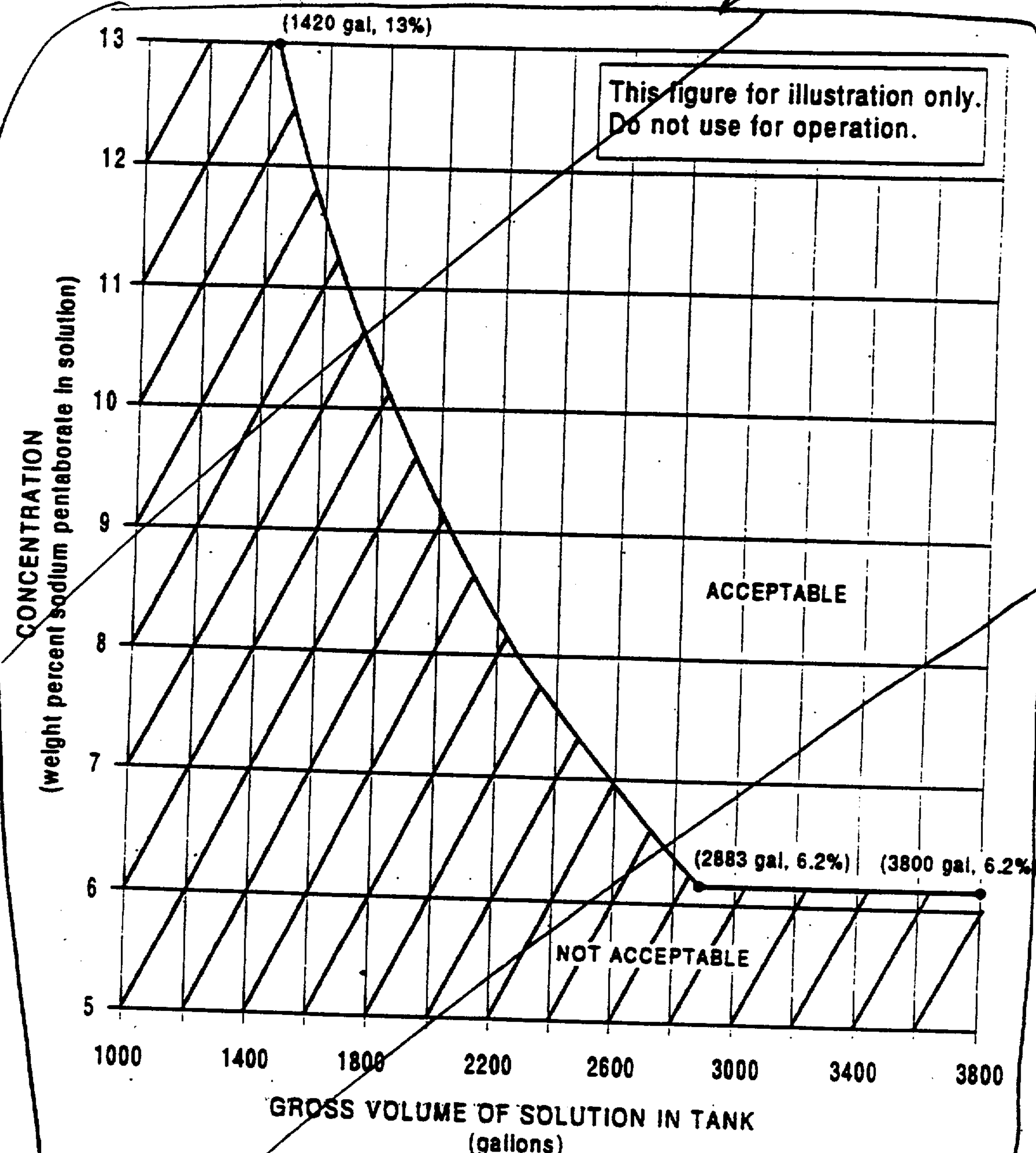


Figure 3.1.7-1 (page 1 of 1)
Sodium Pentaborate Solution Volume
Versus Concentration Requirements

2

⟨CTS⟩

⟨Figure 3.4.A-2⟩

Insert Figure 3.1.7-1

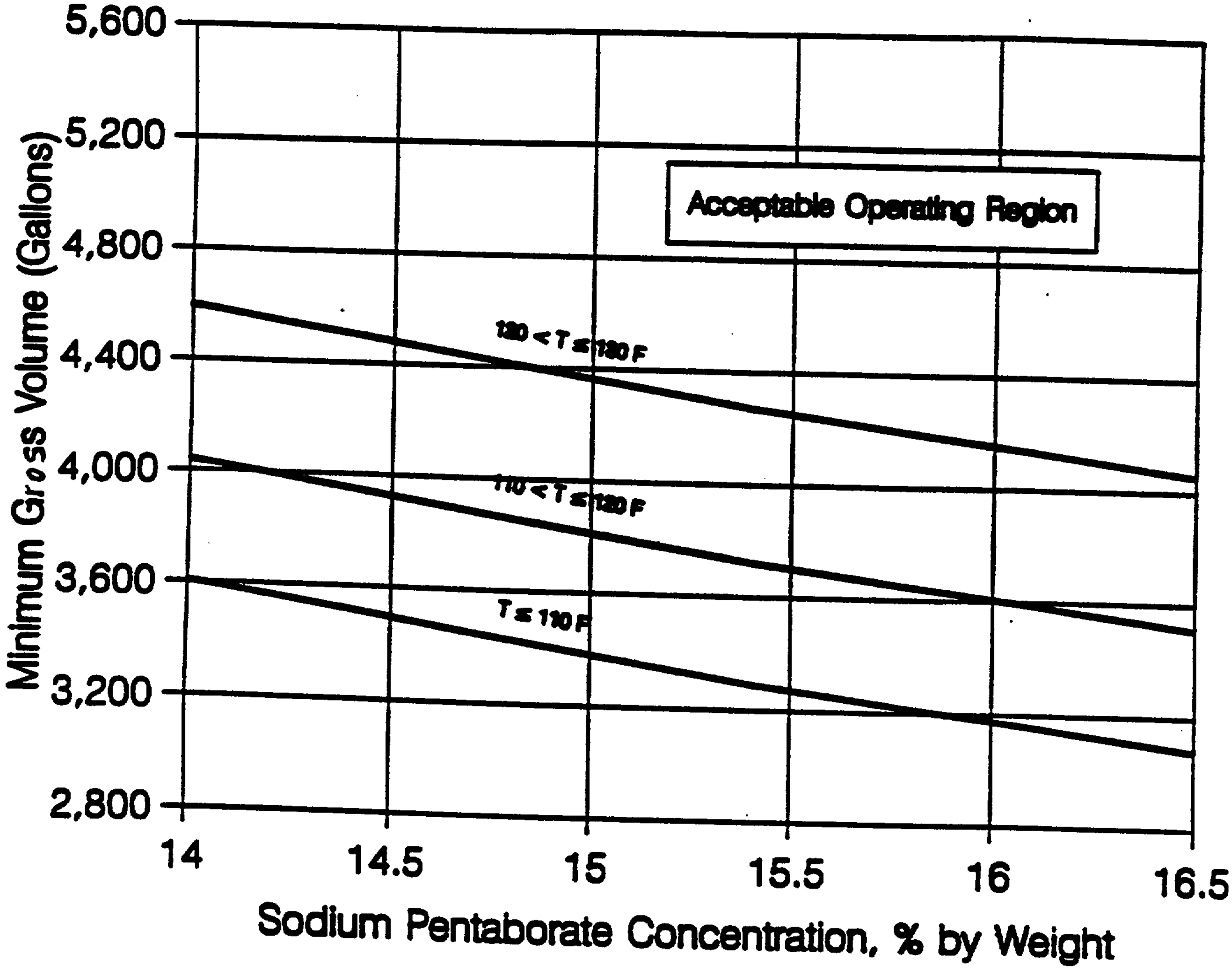


Figure 3.1.7-1 (page 1 of 1)
Sodium Pentaborate Volume Requirements

Insert Figure 3.1.7-2 [2]

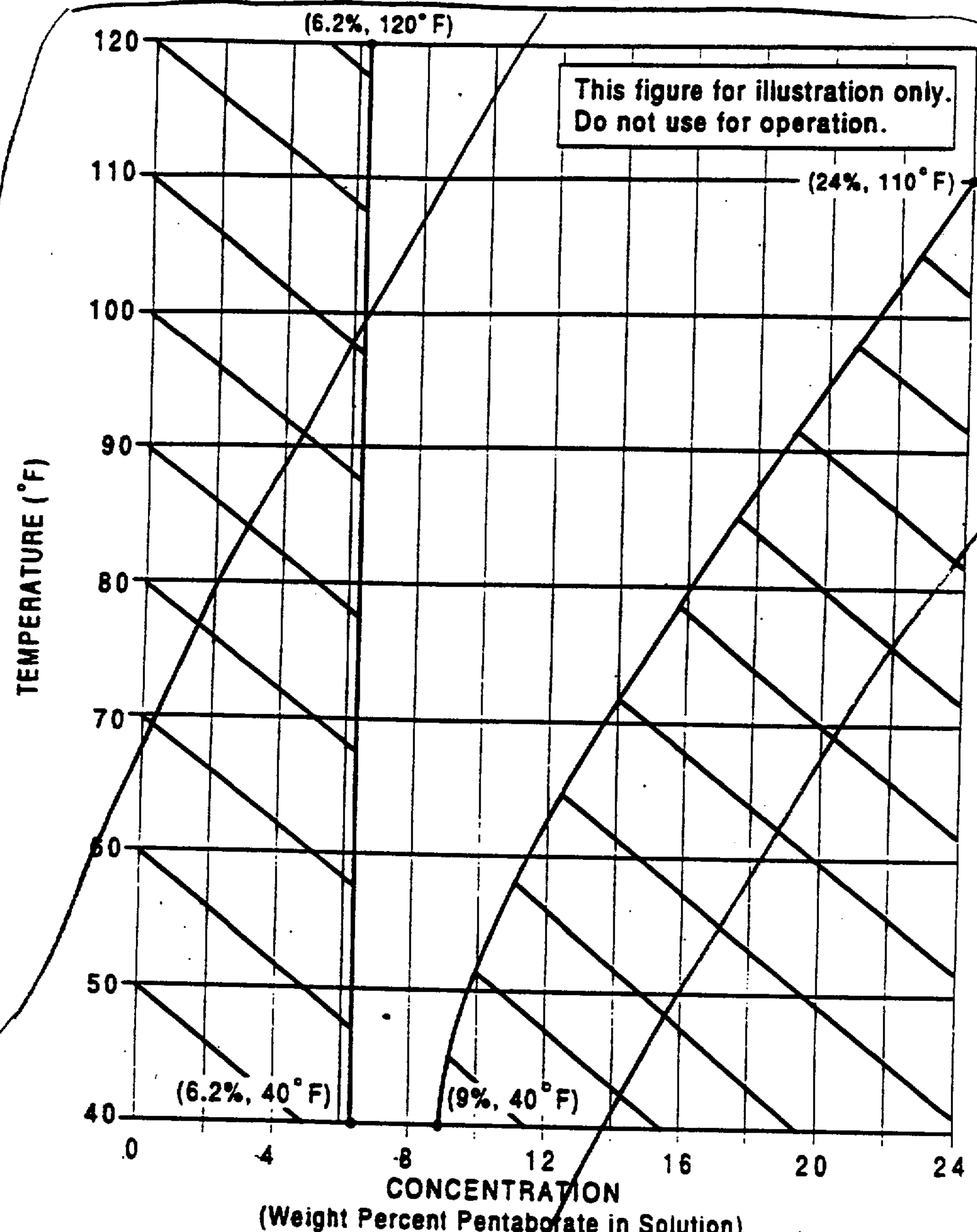


Figure 3.1.7-2 (page 1 of 1)
Sodium Pentaborate Solution Temperature Versus Concentration Requirements

2

<CTS>
<Figure 3.4.A-1>

Insert Figure 3.1.7-2

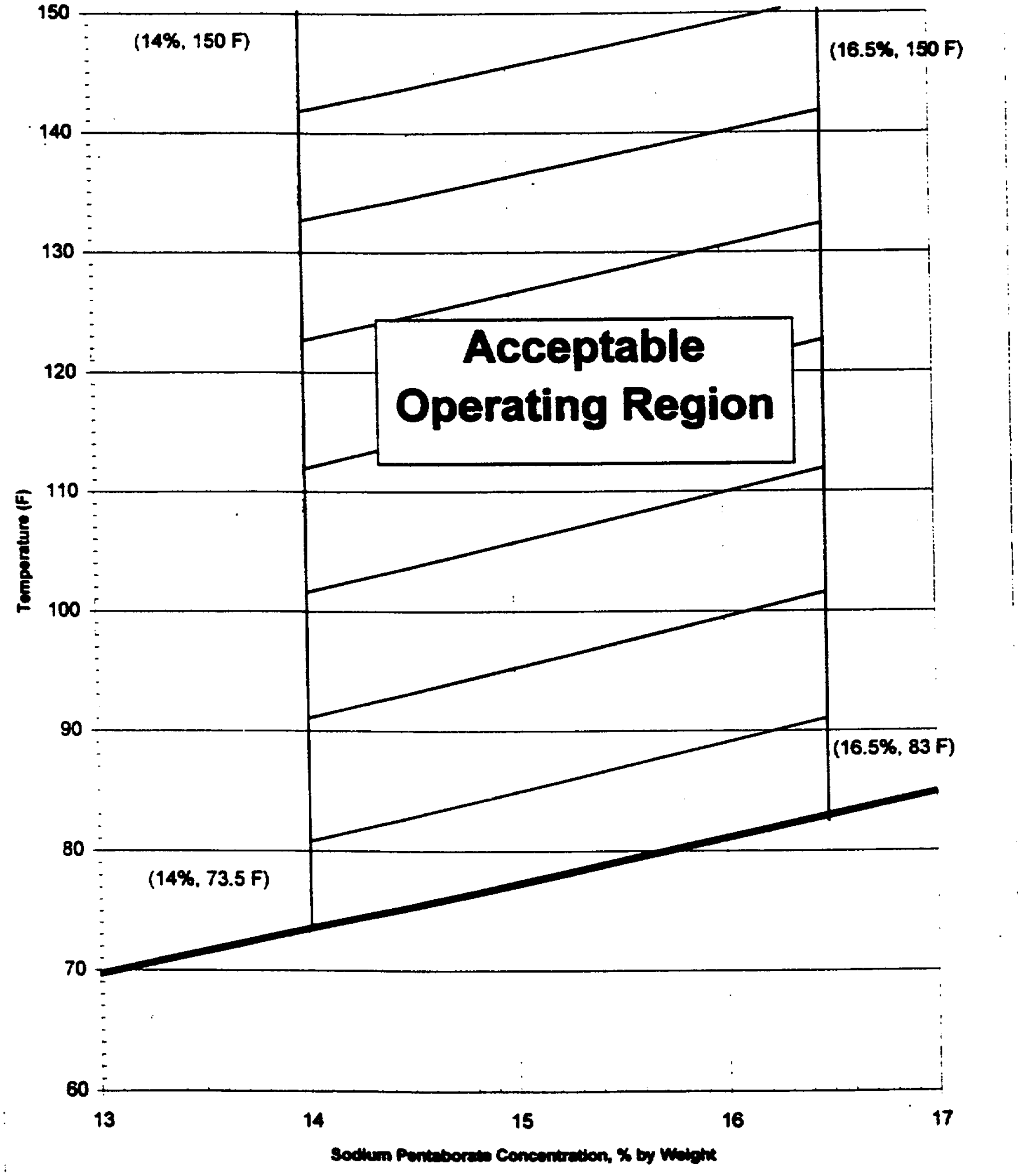


Figure 3.1.7-2 (page 1 of 1)
Sodium Pentaborate Temperature Requirements

1. The bracketed requirement has been deleted since it is not applicable to Quad Cities 1 and 2. The following requirements have been revised and/or renumbered, where applicable, to reflect this deletion. In addition, for SR 3.1.7.6, there are no power operated valves other than the explosive valves (which are tested by other Surveillances), thus this has also been deleted.

2. The brackets have been removed and the proper plant specific information/value has been provided.

3. The proper Quad Cities 1 and 2 nomenclature has been used (CTS Figures 3.4.A-1 and 3.4.A-2). This is also consistent with the nomenclature used in SR 3.1.7.1 and SR 3.1.7.2.

4. The second Frequency for ISTS SR 3.1.7.9 (ITS SR 3.1.7.9) is being changed from being based on solution temperature to piping temperature. The SR requires a verification that all heat traced piping is unblocked. A change in solution temperature in the tank does not necessarily have an impact on the piping temperature, as long as the piping heat trace circuit is functioning properly. The intent of the second Frequency is to ensure that, if the heat tracing is inoperable such that piping temperature falls below the specified minimum temperature, after the heat tracing is restored to OPERABLE status and the piping temperature is greater than or equal to the specified minimum temperature the piping is still unblocked. This is supported by the ISTS Bases description for this second Frequency, which describes the requirement as required to be performed after "piping" temperature is restored.

<CTS>

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

<3.3.K> LCO 3.1.8 Each SDV vent and drain valve shall be OPERABLE.

<Appl 3.3.K> APPLICABILITY: MODES 1 and 2.

1 1.

ACTIONS

5 1

<3.3.K Action footnote b>

-----NOTE-----
Separate Condition entry is allowed for each SDV vent and drain line.

2.

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.3.K Act 1> A. One or more SDV vent or drain lines with one valve inoperable.	A.1 ~~Restore valve to OPERABLE status.~~ Isolate the associated line. 1	7 days
<3.3.K Act 2> B. One or more SDV vent or drain lines with both valves inoperable. <3.3.K Action footnote c>	B.1 -----NOTE----- An isolated line may be unisolated under administrative control to allow draining and venting of the SDV. ---------- 1 Isolate the associated line.	8 hours
<3.3.K Act 1> <3.3.K Act 2> C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
⟨4.3.K.1⟩ SR 3.1.8.1	------NOTE------ Not required to be met on vent and drain valves closed during performance of SR 3.1.8.2. ------------ Verify each SDV vent and drain valve is open.	31 days
⟨4.3.K.2⟩ SR 3.1.8.2	Cycle each SDV vent and drain valve to the fully closed and fully open position.	92 days
⟨4.3.K.3⟩ SR 3.1.8.3	Verify each SDV vent and drain valve: a. Closes in ≤ [60] 30 seconds after receipt of an actual or simulated scram signal; and b. Opens when the actual or simulated scram signal is reset.	~~[18]~~ 24 months 2

1. The ISTS requires that the SDV drain and/or vent valves be restored to operable status if one valve is inoperable. Quad Cities 1 and 2 currently isolates the associated line when one valve is inoperable, instead of requiring the valve to be restored to Operable status. The SDV vent and drain valve's primary function is to isolate the SDV during a scram to contain the reactor coolant discharge. The isolation function can still be satisfied if at least one valve is Operable in each line or the line is isolated. Therefore, Required Action A.1 has been changed to require the associated line to be isolated. In addition, the NOTE of Required Action B.1 has been moved so that it applies to both ACTION A and B. In both cases, it is necessary to unisolate the line under administrative controls to allow draining and venting of the SDV. This is done to prevent the scram on "Scram Discharge Volume Water Level-High." This change has been approved by the NRC in the Safety Evaluation Report for Quad Cities 1 and 2, Amendments 171 and 167, respectively.

2. The brackets have been removed and the proper plant specific information/value has been provided.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

SDM requirements are specified to ensure:

a. The reactor can be made subcritical from all operating conditions and transients and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits; and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

1

UFSAR, Sections 3.1.5 and 4.6.2.1

These requirements are satisfied by the control rods, as described in ~~GDC 26~~ (Ref. 1), which can compensate for the reactivity effects of the fuel and water temperature changes experienced during all operating conditions.

APPLICABLE SAFETY ANALYSES

5

~~The control rod drop accident (CRDA) analysis (Refs. 2 and 3) assumes the core is subcritical with the highest worth control rod withdrawn. Typically, the first control rod withdrawn has a very high reactivity worth and, should the core be critical during the withdrawal of the first control rod, the consequences of a CRDA could exceed the fuel damage limits for a CRDA (see Bases for LCO 3.1.6, "Rod Pattern Control"). Also,~~ SDM is assumed as an initial condition for the control rod removal error during refueling (Ref. 4) ~~and fuel assembly insertion error during refueling (Ref. 5)~~ accidents. The analysis of these reactivity insertion events assumes the refueling interlocks are OPERABLE when the reactor is in the refueling mode of operation. These interlocks prevent the withdrawal of more than one control rod from the core during refueling. (Special consideration and requirements for multiple control rod withdrawal during refueling are covered in Special Operations LCO 3.10.6, "Multiple Control Rod Withdrawal—Refueling.") The analysis assumes this condition is acceptable since the core will be shut down

1 2

15

7 5

(continued)

Having sufficient SDM assures that the reactor will become and remain subcritical after all design basis accidents and transients. For example,

BASES

thereby 2

APPLICABLE SAFETY ANALYSES (continued)

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

positive

Prevention or mitigation of reactivity insertion events is necessary to limit energy deposition in the fuel ~~to~~ prevent ing 2 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.

the 2

1 do

SDM satisfies Criterion 2 of the NRC Policy Statement.

10 CFR 50.36(c)(2)(ii) 1

LCO

The specified SDM limit accounts for the uncertainty in the demonstration of SDM by testing. Separate SDM limits are provided for testing where the highest worth control rod is determined analytically or by measurement. This is due to the reduced uncertainty in the SDM test when the highest worth control rod is determined by measurement. When SDM is demonstrated by calculations not associated with a test (e.g., to confirm SDM during the fuel loading sequence), additional margin is included to account for uncertainties in the calculation. To ensure adequate SDM ~~during the design process~~, a design margin is included to account for uncertainties in the design calculations (Ref. 6). 2

3 1

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)]. 6

to assure shutdown capability 5

(Ref. 2) 1

(continued)

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

2 is available

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

3 2 2

move to page B 3.1.4 as indicated

(continued)

Insert from page B 3.1-3 as indicated 2

BASES

ACTIONS

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

2

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.

Secondary Containment 2

(ensuring components are OPERABLE) 2

Insert ACTION D 2

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.

2 is available

Action must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE; at least one SGT subsystem is OPERABLE; and [secondary containment] isolation capability (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

3 2

Secondary containment 2

move to page B 3.1-5 as indicated

(continued)

2

<u>Insert ACTION D</u>

These administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated.

Insert from page B 3.1-4 as indicated 2

ACTIONS

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

assumed to be isolated to mitigate radioactivity releases. This (ensuring components are OPERABLE) 2 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.

Insert ACTION E 2

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

Adequate SDM must be ~~demonstrated~~ verified 2 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. 2 Adequate SDM is demonstrated by testing before or during the first startup after fuel movement, or 2 control rod replacement, or shuffling within the reactor pressure vessel. 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. ~~2~~ s. 3 and 4 1). 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.

The SDM may be demonstrated during an in - 4 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)

2

Insert ACTION E

These administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated.

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.~~6~~, "Control Rod Testing—Operating"). 6 7

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.

2 During MODES 3 and 4, analytical calculation of SDM may be used to assure the requirements of SR 3.1.1.1 are met.

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

REFERENCES

UFSAR, Sections 3.1.5 and 4.6.2.1 1

1. ~~10 CFR 50, Appendix A, GDC 26~~.
2. ~~FSAR, Section [15.1.38].~~ 1
3. ~~NEDE-24011-P-A-9-US, "General Electric Standard Application for Reactor Fuel," Supplement for United States, Section S.2.2.3.1, September 1988.~~
1 (4) 2 ~~5~~4. FSAR, Section ~~[15.1.23]~~ 15.4.1. 3
5. ~~FSAR, Section [15.1.14].~~

(continued)

BASES 1

REFERENCES (continued)

3 ~~6~~. 4 FSAR, Section ~~14.3.2.~~ 1.3 ~~(2)~~ ~~1~~. 3

4 ~~7~~. NEDE-24011-P-A ~~-9~~, "General Electric Standard Application for Reactor Fuel," ~~Section 3.2.4.1, September 1988~~. 1 as specified in Technical Specification 5.6.5

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

2. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

3. The brackets have been removed and the proper plant specific information has been provided.

4. Typographical/grammatical error corrected.

5. SDM and CRDA analyses are mutually independent in the Quad Cities 1 and 2 reactor safety evaluations. The consideration of SDM is to assure that the reactor is shutdown and remains shutdown with the highest worth control rod withdrawn (and all other control rods inserted). Consequently, the consideration of SDM is no more appropriate for CRDA than it is for other accidents and transients. The CRDA assumes that the highest enthalpy control rod (it is highly probable that this will be different from the highest worth control rod determined for SDM) suddenly drops from the stuck position and falls to the drive position. Doppler reactivity tends to mitigate the event consequences with scram reactivity terminating it.

6. The bracketed information has been deleted since it does not apply.

7. The Bases have been changed to reflect those changes made to the Specifications.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Anomalies

UFSAR, Sections 3.1.5.1, 3.1.5.5, and 3.1.5.6

1

BASES

BACKGROUND

In accordance with GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable such that subcriticality is maintained under cold conditions and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity anomaly is used as a measure of the predicted versus measured core reactivity during power operation. The continual confirmation of core reactivity is necessary to ensure that the Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity anomaly could be the result of unanticipated changes in fuel reactivity or control rod worth or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in assuring the reactor can be brought safely to cold, subcritical conditions. 2

ies

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

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

1

e.g. gadolinia

(continued)

3

BASES

BACKGROUND (continued)

poisons (mainly xenon and samarium) are present in the fuel. # The predicted core reactivity, as represented by ~~control rod density~~ Keffective (keff), is calculated by a 3D core simulator code as a function of cycle exposure. This calculation is performed for projected operating states and conditions throughout the cycle. The core reactivity is determined from ~~control rod densities~~ Keff for actual plant conditions and is then compared to the predicted value for the cycle exposure. 1

APPLICABLE SAFETY ANALYSES

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

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted ~~rod density~~ core keff for identical core conditions at BOC do not reasonably agree, then the assumptions used in the reload cycle design analysis or the calculation models used to predict ~~rod density~~ may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured value. Thereafter, any significant deviations in the measured ~~rod density~~ from the predicted ~~rod density~~ that develop during fuel depletion may be an indication that the assumptions of the DBA and transient analyses are no longer valid, or that an unexpected change in core conditions has occurred. 1

Reactivity anomalies ~~satisfy~~ satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii). 2 1

BASES (continued)

LCO

core keff

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. 1

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. 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, reactivity anomal~~y~~ ies is not required during these conditions. 2

ACTIONS

A.1

Should an anomaly develop between measured and 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)

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.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

core keff

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

1

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1 (continued)

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

1

core keff

The core weight, tons(T) in MWD/T, reflects metric tons

1

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 26~~
2. U FSAR, Chapter ~~[15]~~. 3

1 U

1

UFSAR, Sections 3.1.5.1, 3.1.5.5 and 3.1.5.6

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.1.2 - REACTIVITY ANOMALIES

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

2. Typographical/grammatical error corrected.

3. The brackets have been removed and the proper plant specific information/value has been provided.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Control Rod OPERABILITY

BASES

BACKGROUND

UFSAR, Sections 3.1.5.1, 3.1.5.2 3.1.5.3, 3.1.5.4 3.1.5.5, and 3.1.5.6

1

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

The CRD System consists of ~~137~~ 177 locking piston control rod drive mechanisms (CRDMs) and a hydraulic control unit for each drive mechanism. The locking piston type CRDM is a double acting hydraulic piston, which uses contaminated condensate storage tank, fuel pool reject, or condensate water as the operating fluid. Accumulators provide additional energy for scram. An index tube and piston, coupled to the control rod, are locked at fixed increments by a collet mechanism. The collet fingers engage notches in the index tube to prevent unintentional withdrawal of the control rod, but without restricting insertion.

1

This Specification, along with LCO 3.1.4, "Control Rod Scram Times," and LCO 3.1.5, "Control Rod Scram Accumulators," and LCO 3.1.6, "Rod Pattern Control," ensure that the performance of the control rods in the event of a Design Basis Accident (DBA) or transient meets the assumptions used in the safety analyses of References 2, 3, and 4.

APPLICABLE SAFETY ANALYSES

1 5

The analytical methods and assumptions used in the evaluations involving control rods are presented in References ~~2, 3, and 4~~. The control rods provide the primary means for rapid reactivity control (reactor scram), for maintaining the reactor subcritical and for limiting the potential effects of reactivity insertion events caused by malfunctions in the CRD System.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

The control rods also protect the fuel from damage which could result in release of radioactivity. The limits protected are the MCPR Safety Limit (SL) (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," ~~and~~ LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," and LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoint"), and the fuel ~~damage~~ design limit (see Bases for LCO 3.1.6, "Rod Pattern Control") during reactivity insertion events. 1

The negative reactivity insertion (scram) provided by the CRD System provides the analytical basis for determination of plant thermal limits and provides protection against fuel ~~damage~~ design limits during a CRDA. The Bases for LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6 discuss in more detail how the SLs are protected by the CRD System. 1

Control rod OPERABILITY satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii). 1

LCO

The OPERABILITY of an individual control rod is based on a combination of factors, primarily, the scram insertion times, the control rod coupling integrity, and the ability to determine the control rod position. Accumulator OPERABILITY is addressed by LCO 3.1.5. The associated scram accumulator status for a control rod only affects the scram insertion times; therefore, an inoperable accumulator does not immediately require declaring a control rod inoperable. Although not all control rods are required to be OPERABLE to

(continued)

BASES

LCO (continued)

satisfy the intended reactivity control requirements, strict control over the number and distribution of inoperable control rods is required to satisfy the assumptions of the DBA and transient analyses.

Insert LCO I

APPLICABILITY

In MODES 1 and 2, the control rods are assumed to function during a DBA or transient and are therefore required to be OPERABLE in these MODES. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod OPERABILITY during these conditions. Control rod requirements in MODE 5 are located in LCO 3.9.5, "Control Rod OPERABILITY—Refueling."

ACTIONS

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

A.1, A.2, ~~and~~ A.3, and A.4 TSTF-32

A control rod is considered stuck if it will not insert by either CRD drive water or scram pressure. With a fully inserted control rod stuck, no actions are required as long as the control rod remains fully inserted. The Required Actions are modified by a Note, which allows the rod worth minimizer (RWM) to be bypassed if required to allow continued operation. LCO 3.3.2.1, "Control Rod Block Instrumentation," provides additional requirements when the RWM is bypassed to ensure compliance with the CRDA analysis. With one withdrawn control rod stuck, Insert A-1 TSTF-32 the associated control rod drive must be disarmed in 2 hours. The allowed Completion Time of 2 hours is acceptable, considering the reactor can still be shut down, assuming no additional control rods fail to insert, and provides a reasonable time

(continued)

1

Insert LCO

OPERABILITY requirements for control rods also include correct assembly of the CRD housing supports.

TSTF-32

Insert A-1

the local scram reactivity rate assumptions may not be met if the stuck control rod separation criteria are not met. Therefore, a verification that the separation criteria are met must be performed immediately. The stuck control rod separation criteria are not met if: a) the stuck control rod occupies a location adjacent to two "slow" control rods, b) the stuck control rod is also adjacent to another "slow" control rod, or c) if the stuck control rod occupies a location adjacent to one "slow" control rod when there is another pair of "slow" control rods elsewhere in the core adjacent to one another. The description of "slow" control rods is provided in LCO 3.1.4 "Control Rod Scram Times." In addition, and A.4 TSTF-32

BASES

ACTIONS

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

2

and normal insert and withdraw pressure 2

The control rod must be isolated from both scram and normal insert and withdraw pressure. 2

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. Isolation method should also ensure 2 cooling water to the CRD is maintained. 2

or reactor internals 1

Monitoring of the insertion capability of each withdrawn control rod must also be performed within 24 hours. Insert A-2 TSTF-33 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. TSTF-33 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. 2 ~~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).

Insert A-3 TSTF-33

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). 1

(continued)

TSTF-33

Insert A-2

from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM

TSTF-33

Insert A-3

This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." The Required Action A.3 Completion Time only begins upon discovery of Condition A concurrent with THERMAL POWER greater than

BASES

ACTIONS (continued)

TSTF-34

B.1 ~~and B.2~~ TSTF-34

With two or more withdrawn control rods stuck, ~~the stuck control rods should be isolated from scram pressure within 2 hours and~~ the plant must be 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 to prevent inadvertent withdrawal during subsequent operations. The control rods can be hydraulically disarmed by closing the drive water and exhaust water isolation valves. 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.

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.

(continued)

All changes are $\boxed{3}$ unless otherwise indicated

BASES

ACTIONS (continued)

D.1 and D.2

1

analyzed rod position

s. 6 and 7

Out of sequence control rods may increase the potential reactivity worth of a dropped control rod during a CRDA. At ≤ 10% RTP, the ~~generic banked position withdrawal~~ sequence ~~(BPWS)~~ analysis (Ref. 5) requires inserted control rods not in compliance with BPWS to be separated by at least two OPERABLE control rods in all directions, including the diagonal. Therefore, if two or more inoperable control rods are not in compliance with BPWS and not separated by at least two OPERABLE control rods, action must be taken to restore compliance with BPWS or restore the control rods to OPERABLE status. Condition D is modified by a Note indicating that the Condition is not applicable when > 10% RTP, since the BPWS is not required to be followed under these conditions, as described in the Bases for LCO 3.1.6. The allowed Completion Time of 4 hours is acceptable, considering the low probability of a CRDA occurring.

the analyzed rod position sequence

2

(i.e., all other control rods in a five by five array centered on the inoperable control rod are OPERABLE)

in all directions

2

analyzed rod position sequence

~~E.1~~

~~In addition to the separation requirements for inoperable control rods, an assumption in the CRDA analysis for ANF fuel is that no more than three inoperable control rods are allowed in any one BPWS group. Therefore, with one or more BPWS groups having four or more inoperable control rods, the control rods must be restored to OPERABLE status. Required Action E.1 is modified by a Note indicating that the Condition is not applicable when THERMAL POWER is > 10% RTP since the BPWS is not required to be followed under these conditions, as described in the Bases for LCO 3.1.6. The allowed Completion Time of 4 hours is acceptable, considering the low probability of a CRDA occurring.~~

3

3

E F.1

If any Required Action and associated Completion Time of Condition A, C, D, or E are not met, or there are nine or more inoperable control rods, 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. This ensures all insertable control rods are inserted and places the reactor in a condition that does not require the

(continued)

BASES

3

ACTIONS

E ~~D~~.1 (continued)

active function (i.e., scram) of the control rods. The number of control rods permitted to be inoperable when operating above 10% RTP (e.g., no CRDA considerations) could be more than the value specified, but the occurrence of a large number of inoperable control rods could be indicative of a generic problem, and investigation and resolution of the potential problem should be undertaken. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

control rod 4

The position of each control rod must be determined to ensure adequate information on control rod position is available to the operator for determining ~~CRD~~ OPERABILITY and controlling rod patterns. Control rod position may be determined by the use of OPERABLE position indicators, by moving control rods to a position with an OPERABLE indicator, or by the use of other appropriate methods. The 24 hour Frequency of this SR is based on operating experience related to expected changes in control rod position and the availability of control rod position indications in the control room.

1

(full-in, full-out, or numeric indicators)

by verifying the indicators one notch "out" and one notch "in" are OPERABLE,

SR 3.1.3.2 and SR 3.1.3.3

analyzed rod position sequence

Control rod insertion capability is demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. The control rod may then be returned to its original position. This ensures the control rod is not stuck and is free to insert on a scram signal. These Surveillances are not required when THERMAL POWER is less than or equal to the actual LPSP of the RWM, since the notch insertions may not be compatible with the requirements of the ~~Banked Position Withdrawal Sequence (BPWS)~~ (LCO 3.1.6) and the RWM (LCO 3.3.2.1). The 7 day Frequency of SR 3.1.3.2 is based on operating experience related to the changes in CRD performance and the ease of performing notch testing for fully withdrawn control rods. Partially withdrawn control

2

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.3.2 and SR 3.1.3.3 (continued)

rods are tested at a 31 day Frequency, based on the potential power reduction required to allow the control rod movement and considering the large testing sample of SR 3.1.3.2. Furthermore, the 31 day Frequency takes into account operating experience related to changes in CRD performance. At any time, if a control rod is immovable, a determination of that control rod's trippability (OPERABILITY) must be made and appropriate action taken.

2 Insert SR 3.1.3.2 and SR 3.1.3.3

SR 3.1.3.4

Verifying that the scram time for each control rod to notch position 06 is ≤ 7 seconds provides reasonable assurance that the control rod will insert when required during a DBA or transient, thereby completing its shutdown function. This SR is performed in conjunction with the control rod scram time testing of SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," and the functional testing of SDV vent a drain valves in LCO 3.1.8, "Scram Discharge Volume (SDV) Vent and Drain Valves," overlap this Surveillance to provide complete testing of the assumed safety function. The associated Frequencies are acceptable, considering the more frequent testing performed to demonstrate other aspects of control rod OPERABILITY and operating experience, which shows scram times do not significantly change over an operating cycle.

90% insertion 3

SR 3.1.3.5

Coupling verification is performed to ensure the control rod is connected to the CRDM and will perform its intended function when necessary. The Surveillance requires verifying that a control rod does not go to the withdrawn overtravel position when it is fully withdrawn. The overtravel position feature provides a positive check on the coupling integrity since only an uncoupled CRD can reach the overtravel position. The verification is required to be performed any time a control rod is withdrawn to the "full out" position (notch position 48) or prior to declaring the control rod OPERABLE after work on the control rod or CRD System that could

2

(continue

2

<u>Insert SR 3.1.3.2 and SR 3.1.3.3</u>

These SRs are modified by Notes that allow 7 days and 31 days respectively, after withdrawal of the control rod and increasing power to above the LPSP, to perform the Surveillance. This acknowledges that the control rod must be first withdrawn and THERMAL POWER must be increased to above the LPSP before performance of the Surveillance, and therefore, the Notes avoid potential conflicts with SR 3.0.3 and SR 3.0.4.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.3.5 (continued)

affect coupling. This includes control rods inserted one notch and then returned to the "full out" position during the performance of SR 3.1.3.2. This Frequency is acceptable, considering the low probability that a control rod will become uncoupled when it is not being moved and operating experience related to uncoupling events.

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 26, GDC 27, GDC 28, and GDC 29.~~ UFSAR, Sections 3.1.5.1, 3.1.5.2, 3.1.5.3, 3.1.5.4, 3.1.5.5 and 3.1.5.6 [1]

2. UFSAR, Section ~~[4.2.3.2.2.4]~~ 5.2.2.2.3.

3. UFSAR, Section ~~[5X.4.3]~~ 6.2.1.3.2. [5]

4. UFSAR, ~~Section [15.1]~~ Chapter 15. [5]

5. UFSAR, Section 4.6.3.4.2.1 [1]

6. NEDO-21231, "Banked Position Withdrawal Sequence," Section 7.2, January 1977. [1]

7. NFSR-0091, Commonwealth Edison Topical Report, Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods, (as specified in Technical Specification 5.6.5)

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.1.3 - CONTROL ROD OPERABILITY

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

2. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

3. Changes have been made to reflect those changes made to the Specification.

4. Typographical/grammatical error corrected.

5. The brackets have been removed and the proper plant specific information/value has been provided.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Control Rod Scram Times

BASES

[1]

BACKGROUND

The scram function of the Control Rod Drive (CRD) System controls reactivity changes during ~~abnormal operational transients~~ anticipated operational occurrences to ensure that specified acceptable fuel design limits are not exceeded (Ref. 1). The control rods are scrammed by positive means using hydraulic pressure exerted on the CRD piston.

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

[1]

APPLICABLE SAFETY ANALYSES

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

(continued)

BASES

2

At ≥

APPLICABLE SAFETY ANALYSES (continued)

The scram function of the CRD System protects the MCPR Safety Limit (SL) (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") ~~and~~ the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR),"), which ensure that no fuel damage will occur if these limits are not exceeded. Above 800 psig, the scram function is designed to insert negative reactivity at a rate fast enough to prevent the actual MCPR from becoming less than the MCPR SL, during the analyzed limiting power transient. Below 800 psig, the scram function is assumed to perform during the control rod drop accident (Ref. 6) and, therefore, also provides protection against violating fuel ~~damage~~ limits during reactivity insertion accidents (see Bases for LCO 3.1.6, "Rod Pattern Control"). For the reactor vessel overpressure protection analysis, the scram function, along with the safety/relief valves, ensure that the peak vessel pressure is maintained within the applicable ASME Code limits.

LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," and LCO 3.2.4 "Average Power Range Monitor (APRM) Gain and Setpoint"

3 1

design 1

Control rod scram times satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (c)(2)(ii) 1

LCO

The scram times specified in Table 3.1.4-1 ~~(in the accompanying LCO)~~ are required to ensure that the scram reactivity assumed in the DBA and transient analysis is met (Ref. 6). To account for single failures and "slow" scramming control rods, the scram times specified in Table 3.1.4-1 are faster than those assumed in the design basis analysis. The scram times have a margin that allows up to approximately 7% of the control rods (e.g., 137 x 7% = 10) to have scram times exceeding the specified limits (i.e., "slow" control rods) assuming a single stuck control rod (as allowed by LCO 3.1.3, "Control Rod OPERABILITY") and an additional control rod failing to scram per the single failure criterion. The scram times are specified as a function of reactor steam dome pressure to account for the pressure dependence of the scram times. The scram times are specified relative to measurements based on reed switch positions, which provide the control rod position indication. The reed switch closes ("pickup") when the index tube passes a specific location and then opens ("dropout") as the index tube travels upward. Verification of the specified scram times in Table 3.1.4-1 is accomplished

2

4

12

177

1

(continued)

and interpolation of the "pickup" or "dropout" times of reed switches associated with each of the required insertion positions 1

(face or diagonal) 2

BASES

LCO (continued)

through measurement ~~of the "dropout" times~~. To ensure that local scram reactivity rates are maintained within acceptable limits, no more than two of the allowed "slow" control rods may occupy adjacent locations.

Table 3.1.4-1 is modified by two Notes which state that control rods with scram times not within the limits of the table are considered "slow" and that control rods with scram times > 7 seconds are considered inoperable as required by SR 3.1.3.4.

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

APPLICABILITY

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

ACTIONS

A.1

When the requirements of this LCO are not met, the rate of negative reactivity insertion during a scram may not be within the assumptions of the safety analyses. Therefore, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

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

SR 3.1.4.1

The scram reactivity used in DBA and transient analyses is based on an assumed control rod scram time. Measurement of the scram times with reactor steam dome pressure ≥ 800 psig demonstrates acceptable scram times for the transients analyzed in References ~~3 and 4~~ 5, 6 and 7. [7]

Maximum scram insertion times occur at a reactor steam dome pressure of approximately 800 psig because of the competing effects of reactor steam dome pressure and stored accumulator energy. Therefore, demonstration of adequate scram times at reactor steam dome pressure ≥ 800 psig ensures that the measured scram times will be within the specified limits at higher pressures. Limits are specified as a function of reactor pressure to account for the sensitivity of the scram insertion times with pressure and to allow a range of pressures over which scram time testing can be performed. To ensure that scram time testing is performed within a reasonable time following ~~fuel movement within the reactor pressure vessel after~~ a shutdown ≥ 120 days or longer, control rods are required to be tested before exceeding 40% RTP following the shutdown. ~~In the event fuel movement is limited to selected core cells, it is the intent of this SR that only those CRDs associated with the core cells affected by the fuel movements are required to be scram time tested. However, if the reactor remains shutdown ≥ 120 days, all control rods are required to be scram time tested.~~ This Frequency is acceptable considering the additional surveillances performed for control rod OPERABILITY, the frequent verification of adequate accumulator pressure, and the required testing of control rods affected by fuel movement within the associated core cell and by work on control rods or the CRD System. [7] TSTF-222

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.2

3

i.e

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

2

scram time

found in the Technical Requirements Manual (Ref. 8) and are

2

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 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."

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.3 (continued)

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

or when fuel movement within the reactor pressure vessel occurs, TSTF-222

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.

TSTF-222

When fuel movement within the reactor pressure vessel occurs, only those control rods associated with the core cells affected by the fuel movement are required to be scram time tested. During a routine refueling outage, it is expected that all control rods will be affected.

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

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 10~~ UFSAR, Section 3.1 [1]
2. (U)FSAR, Section ~~4.2.3.2.2.4~~ 4.6.3.4.2.1 [4] [1]
3. ~~FSAR, Section [5A.4.3]~~ [1]

(continued)

BASES

REFERENCES (continued)

~~4.~~ 3. UFSAR, Section ~~15.2~~ 4.10.

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

~~6.~~ 4. 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.

5. UFSAR, Section 5.2.2.2.3.

6. UFSAR, Section 6.2.1.3.2.

7. UFSAR, Chapter 15.

8. Technical Requirements Manual.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.1.4 - CONTROL ROD SCRAM TIMES

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

2. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

3. Typographical/grammatical error corrected.

4. The brackets have been removed and the proper plant specific information/value has been provided.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Control Rod Scram Accumulators

BASES

BACKGROUND

The control rod scram accumulators are part of the Control Rod Drive (CRD) System and are provided to ensure that the control rods scram under varying reactor conditions. The control rod scram accumulators store sufficient energy to fully insert a control rod at any reactor vessel pressure. The accumulator is a hydraulic cylinder with a free floating piston. The piston separates the water used to scram the control rods from the nitrogen, which provides the required energy. The scram accumulators are necessary to scram the control rods within the required insertion times of LCO 3.1.4, "Control Rod Scram Times."

APPLICABLE SAFETY ANALYSES

The analytical ~~methods and~~ assumptions used in evaluating the control ~~rod scram~~ function are presented in Reference~~s~~ 1, ~~2, and 3~~. The Design Basis Accident (DBA) and transient analyses assume that all of the control rods scram at a specified insertion rate. OPERABILITY of each individual control rod scram accumulator, along with LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.4, ensures that the scram reactivity assumed in the DBA and transient analyses can be met. The existence of an inoperable accumulator may invalidate prior scram time measurements for the associated control rod.

1

The scram function of the CRD System, and therefore the OPERABILITY of the accumulators, protects the MCPR Safety Limit (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," ~~and~~ LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)" and LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoint"), which ensure that no fuel damage will occur if these limits are not exceeded (see Bases for LCO 3.1.4). In addition, the scram function at low reactor vessel pressure (i.e., startup conditions) provides protection against violating fuel design limits during reactivity insertion accidents (see Bases for LCO 3.1.6, "Rod Pattern Control").

1

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	Control rod scram accumulators satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii). 1
LCO	The OPERABILITY of the control rod scram accumulators is required to ensure that adequate scram insertion capability exists when needed over the entire range of reactor pressures. The OPERABILITY of the scram accumulators is based on maintaining adequate accumulator pressure.
APPLICABILITY	In MODES 1 and 2, the scram function is required for mitigation of DBAs and transients, and therefore the scram accumulators must be OPERABLE to support the scram function. In MODES 3 and 4, control rods are ~~only allowed~~ not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod scram accumulator OPERABILITY during these conditions. Requirements for scram accumulators in MODE 5 are contained in LCO 3.9.5, "Control Rod OPERABILITY—Refueling." 2
ACTIONS	The ACTIONS ~~t~~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 ~~affected~~ inoperable accumulator. Complying with the Required Actions may allow for continued operation and subsequent ~~affected~~ inoperable accumulators governed by subsequent Condition entry and application of associated Required Actions. 3 2 2

A.1 and A.2

With one control rod scram accumulator inoperable and the reactor steam dome pressure ≥ 900 psig, 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 in Table 3.1.4-1.

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(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

3 3 rod 2 surveillance may 3

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

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

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

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

3 rod

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

surveillance 2

(continued)

BASES

ACTIONS

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

may 3 ~~would~~ already be considered "slow" and the further degradation of scram performance with an inoperable accumulator could result in excessive scram times. In this event, the associated control rod is declared inoperable (Required Action B.2.2) and LCO 3.1.3 entered. This would result in requiring the affected control rod to be fully inserted and disarmed, thereby satisfying its intended function in accordance with ACTIONS of LCO 3.1.3.

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

C.1 and C.2

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

D.1

The reactor mode switch must be immediately placed in the shutdown position if either Required Action and associated Completion Time associated with loss of the CRD ~~charging~~ 1 pump (Required Actions B.1 and C.1) cannot be met. This

(continued)

BASES

ACTIONS

D.1 (continued)

ensures that all insertable control rods are inserted and that the reactor is in a condition that does not require the active function (i.e., scram) of the control rods. This Required Action is modified by a Note stating that the action is not applicable if all control rods associated with the inoperable scram accumulators are fully inserted, since the function of the control rods has been performed.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

SR 3.1.5.1 requires that the accumulator pressure be checked every 7 days to ensure adequate accumulator pressure exists to provide sufficient scram force. The primary indicator of accumulator OPERABILITY is the accumulator pressure. A minimum accumulator pressure is specified, below which the capability of the accumulator to perform its intended function becomes degraded and the accumulator is considered inoperable. The minimum accumulator pressure of 940 psig is well below the expected pressure of 1100 psig (Ref. ~~3~~ 2). Declaring the accumulator inoperable when the minimum pressure is not maintained ensures that significant degradation in scram times does not occur. The 7 day Frequency has been shown to be acceptable through operating experience and takes into account indications available in the control room. 2 1

REFERENCES u 1

1. u FSAR, Section [~~4.2.3.2.2.4~~]. 4.6.3.4.2.1 4

~~2. FSAR, Section [5A.4.3].~~ 1

~~3. FSAR, Section [15.1].~~

2. Letter, from E. Y. Gibo (GE) to P. Chenell (ComEd), "Generic Basis for HCU Scram Accumulator Minimum Setpoint Pressure," April 10 1998. 1

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.1.5 - CONTROL ROD SCRAM ACCUMULATORS

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

2. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

3. Typographical/grammatical error corrected.

4. The brackets have been removed and the proper plant specific information/value has been provided.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Rod Pattern Control

BASES

BACKGROUND

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

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

APPLICABLE SAFETY ANALYSES

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

Prevention or mitigation of positive reactivity insertion events is necessary to limit the energy deposition in the fuel, thereby preventing significant fuel damage which could result in the undue release of radioactivity. Since the failure consequences for UO_2 have been shown to be insignificant below fuel energy depositions of 300 cal/gm (Ref. 3), the fuel damage limit of 280 cal/gm provides a margin of safety from significant core damage which would result in release of radioactivity (Refs. 4 and 5). Generic evaluations (Refs. 1 and 6) of a design basis CRDA (i.e., a CRDA resulting in a peak fuel energy deposition of 280 cal/gm) have shown that if the peak fuel enthalpy remains below 280 cal/gm, then the maximum reactor pressure will be less than the required ASME Code limits (Ref. 7) and the calculated offsite doses will be well within the required limits (Ref. 8).

(continued)

Cycle specific CRDA analyses are performed that assume eight inoperable control rods with at least two cell separation and confirm fuel energy deposition is less than 280 cal/gm.

All changes are 2 unless otherwise indicated

predetermined sequencing rules (analyzed rod position sequence)

the cycle specific analyses

analyzed rod position sequence

BASES

APPLICABLE SAFETY ANALYSES (continued)

Control rod patterns analyzed in ~~Reference~~ follow ~~the banked position withdrawal sequence (BPWS)~~. The ~~BPWS~~ is applicable from the condition of all control rods fully inserted to 10% RTP (Ref. 2). ~~For the BPWS~~ the control rods are required to be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions (e.g., between notches 08 and 12). The banked positions are established to minimize the maximum incremental control rod worth without being overly restrictive during normal plant operation. ~~Generic analysis of the BPWS (Ref. 1) has demonstrated~~ that the 280 cal/gm fuel ~~damage~~ limit will not be violated during a CRDA ~~while following the BPWS MODE of operation~~. The ~~generic BPWS analysis~~ (Ref. 8) also evaluated the effect of fully inserted, inoperable control rods not in compliance with the sequence, to allow a limited number (i.e., eight) and distribution of fully inserted, inoperable control rods.

5 | 1 | cycle specific analyses ensure | under worst case scenarios | design | 1,2,3,4 and 5 | 5 | cycle specific analyses

Specific analysis may also be performed for atypical operating conditions (e.g. fuel leaker suppression).

Rod pattern control satisfies Criterion 3 of ~~the NRC Policy Statement~~. 10 CFR 50.36(c)(2)(ii)

LCO

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

analyzed rod position sequence

APPLICABILITY

In MODES 1 and 2, when THERMAL POWER is ≤ 10% RTP, the CRDA is a Design Basis Accident and, therefore, compliance with the assumptions of the safety analysis is required. When THERMAL POWER is > 10% RTP, there is no credible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel ~~damage~~ limit during a CRDA (Ref. 2). In MODES ~~3, 4, and~~ 5, since the reactor is shut down and only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will remain subcritical with a single control rod withdrawn.

1 | design | 4 and 5 | 5 | 5

(continued)

5

In MODES 3 and 4, the reactor is shutdown and the control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied, therefore a CRDA is not postulated to occur

BASES (continued)

ACTIONS

A.1 and A.2

With one or more OPERABLE control rods not in compliance with the prescribed control rod sequence, actions may be taken to either correct the control rod pattern or declare the associated control rods inoperable within 8 hours. Noncompliance with the prescribed sequence may be the result of "double notching," drifting from a control rod drive cooling water transient, leaking scram valves, or a power reduction to ≤ ~~10%~~ RTP before establishing the correct control rod pattern. The number of OPERABLE control rods not in compliance with the prescribed sequence is limited to eight, to prevent the operator from attempting to correct a control rod pattern that significantly deviates from the prescribed sequence. When the control rod pattern is not in compliance with the prescribed sequence, all control rod movement should be stopped except for moves needed to correct the rod pattern, or scram if warranted. 1 3

second licensed operator (Reactor Operator or Senior Reactor Operator) or by a task 4

Required Action A.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position. LCO 3.3.2.1 requires verification of control rod movement by a qualified member of the technical staff. This ensures that the control rods will be moved to the correct position. A control rod not in compliance with the prescribed sequence is not considered inoperable except as required by Required Action A.2. OPERABILITY of control rods is determined by compliance with LCO 3.1.3, "Control Rod OPERABILITY," LCO 3.1.4, "Control Rod Scram Times," and LCO 3.1.5, "Control Rod Scram Accumulators." The allowed Completion Time of 8 hours is reasonable, considering the restrictions on the number of allowed out of sequence control rods and the low probability of a CRDA occurring during the time the control rods are out of sequence.

helps to 5

6

5

(e.g., a shift technical advisor or reactor engineer)

B.1 and B.2

If nine or more OPERABLE control rods are out of sequence, the control rod pattern significantly deviates from the prescribed sequence. Control rod withdrawal should be suspended immediately to prevent the potential for further deviation from the prescribed sequence. Control rod insertion to correct control rod withdrawn beyond their allowed position is allowed since, in general, insertion of

(continued)

BASES

5

ACTIONS

B.1 and B.2 (continued)

(e.g., a shift technical advisor or reactor engineer)

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 qualified member of the technical staff.

second licensed operator (Reactor Operator or Senior Reactor Operator) or by a task

4

the analyzed rod position sequence

2

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.

analyzed rod position sequence

2

1

REFERENCES

Insert Ref-1

2

1. ~~NEDE-24011-P-A-9-US, "General Electric Standard Application for Reactor Fuel, Supplement for United States," Section 2.2.3.1, September 1988.~~

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

2

6 ~~3~~. NUREG-0979, Section 4.2.1.3.2, April 1983.

7 ~~4~~. NUREG-0800, Section 15.4.9, Revision 2, July 1981.

(continued)

2

Insert Ref-1

1. UFSAR, Section 15.4.10.

2. XN-NF-80-19(P)(A), Volume 1, Supplement 2, Section 7.1 Exxon Nuclear Methodology for Boiling Water Reactor-Neutronics Methods for Design and Analysis, (as specified in Technical Specification 5.6.5).

3. NEDE-24011-P-A, "GE Standard Application for Reactor Fuel," (as specified in Technical Specification 5.6.5).

4. Letter from T.A. Pickens (BWROG) to G.C. Lainas (NRC), "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A," BWROG-8644, August 15, 1986.

5. NFSR-0091, Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods, Commonwealth Edison Topical Report, (as specified in Technical Specification 5.6.5).

2

BASES

REFERENCES (continued)

11 ~~5.~~ 10 CFR 100.11.

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

10 ~~7.~~ ASME, Boiler and Pressure Vessel Code.

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

9. NEDO-10527 "Rod Drop Accident Analysis for Large BWRs," (including Supplements 1 and 2), March 1972.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.1.6 - ROD PATTERN CONTROL

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

3. This requirement has been deleted since the ACTIONS do not require that all rod movement (except for the moves needed to correct the rod pattern or a scram) be suspended.

4. Changes have been made to more clearly match the requirements of the LCO.

5. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

6. A 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 ITS 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.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

BASES

BACKGROUND

The SLC System is designed to provide the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory (which is at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive, xenon free state without taking credit for control rod movement. The SLC System satisfies the requirements of 10 CFR 50.62 (Ref. 1) on anticipated transient without scram.

The SLC System consists of a boron solution storage tank, two positive displacement pumps, two explosive valves that are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged near the bottom of the core shroud, where it then mixes with the cooling water rising through the core. A smaller tank containing demineralized water is provided for testing purposes.

7 determines

APPLICABLE SAFETY ANALYSES

The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator ~~believes~~ the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System is used in the event that enough control rods cannot be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to add negative reactivity to compensate for all of the various reactivity effects that could occur during plant operations. To meet this objective, it is necessary to inject a quantity of boron, which produces a concentration of ~~660~~ 600 ppm of natural boron, in the reactor coolant at 68°F. To allow for potential leakage and imperfect mixing in the reactor system, an amount of boron equal to 25% of the amount cited above is added (Ref. 2). The volume versus concentration limits in Figure 3.1.7-1 and the temperature versus concentration limits in Figure 3.1.7-2 are calculated such that the required concentration is achieved accounting for dilution in the RPV with ~~normal~~ reactor water level ~~and~~ at the high alarm point, including the water volume in

1

(continued)

[1] and portions of other piping systems which connect to the RPV below the high alarm point

BASES

APPLICABLE SAFETY ANALYSES (continued)

the residual heat removal shutdown cooling piping ~~and in~~ the recirculation loop piping. This quantity of borated solution is the amount that is above the ~~pump suction shutoff level in~~ the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected.

9 [1] bottom of [1] represented However

The SLC System satisfies ~~the requirements~~ of ~~the NRC Policy Statement because operating experience and probabilistic risk assessments have shown the SLC System to be important to public health and safety. Thus, it is retained in the Technical Specifications~~.

Criterion 4 10 CFR 50.36(c)(2)(ii) [2]

LCO

The OPERABILITY of the SLC System provides backup capability for reactivity control independent of normal reactivity control provisions provided by the control rods. The OPERABILITY of the SLC System is based on the conditions of the borated solution in the storage tank and the availability of a flow path to the RPV, including the OPERABILITY of the pumps and valves. Two SLC subsystems are required to be OPERABLE; each contains an OPERABLE pump, an explosive valve, and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path.

[7] Insert LCO

APPLICABILITY

In MODES 1 and 2, shutdown capability is required. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate controls to ensure that the reactor remains subcritical. In MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Demonstration of adequate SDM (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") ensures that the reactor will not become critical. Therefore, the SLC System is not required to be OPERABLE when only a single control rod can be withdrawn.

ACTIONS

~~A.1~~

~~If the boron solution concentration is less than the required limits for mitigation but greater than the~~

[3]

(continued)

7

Insert LCO

With one subsystem inoperable the requirements of 10 CFR 50.62 (Ref. 1) cannot be met, however, the remaining subsystem is still capable of shutting down the unit.

BASES

ACTIONS

3

A.1 (continued)

concentration required for cold shutdown (original licensing basis), the concentration must be restored to within limits in 72 hours. It is not necessary under these conditions to enter Condition C for both SLC subsystems inoperable since they are capable of performing their original design basis function. Because of the low probability of an event and the fact that the SLC System capability still exists for vessel injection under these conditions, the allowed Completion Time of 72 hours is acceptable and provides adequate time to restore concentration to within limits.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of concentration out of limits or inoperable SLC subsystems during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, an SLC subsystem is inoperable and that subsystem is subsequently returned the OPERABLE, the LCO may already have been not met for up to 7 days. This situation could lead to a total duration of 10 days (7 days in Condition B, followed by 3 days in Condition A), since initial failure of the LCO, to restore the SLC System. Then an SLC subsystem could be found inoperable again, and concentration could be restored to within limits. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock," resulting in establishing the "time zero" at the time the LCO was initially not met instead of at the time Condition A was entered. The 10 day Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

3

A ~~B~~.1

3 3

If one SLC subsystem is inoperable for reasons other than Condition A, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE subsystem is adequate to ~~perform the~~ shutdown the unit ~~function~~. However, the overall ~~reliability~~ capability is reduced since ~~because a single failure in~~ the remaining OPERABLE subsystem ~~could result in reduced SLC System shutdown~~ cannot meet the requirements of Reference 1.

3

(continued)

BASES

ACTIONS

A 3

~~B~~.1 (continued)

~~capability.~~ The 7 day Completion Time is based on the availability of an OPERABLE subsystem capable of ~~performing the intended SLC System function~~ Shutting down the reactor and the low probability of a Design Basis Accident (DBA) or severe transient occurring concurrent with the failure of the Control Rod Drive (CRD) System to shut down the ~~plant~~ reactor. 3 1

~~The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of concentration out of limits or inoperable SLC subsystems during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, concentration is out of limits, and is subsequently returned to within limits, the LCO may already have been not met for up to 3 days. This situation could lead to a total duration of 10 days (3 days in Condition A, followed by 7 days in Condition B), since initial failure of the LCO, to restore the SLC System. Then concentration could be found out of limits again, and the SLC subsystem could be restored to OPERABLE. This could continue indefinitely.~~ 3

~~This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock," resulting in establishing the "time zero" at the time the LCO was initially not met instead of at the time Condition B was entered. The 10 day Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.~~

3

B ~~C~~.1

If both SLC subsystems are inoperable ~~for reasons other than Condition A~~, at least one subsystem must be restored to OPERABLE status within 8 hours. The allowed Completion Time of 8 hours is considered acceptable given the low probability of a DBA or transient occurring concurrent with the failure of the control rods to shut down the reactor. 3

(continued)

BASES

ACTIONS (continued)

C.1 E.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

SR 3.1.7.1 through SR 3.1.7.3 are 24 hour Surveillances 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 Surveillances ensure that the proper borated solution volume and temperature, including the temperature of the pump suction piping, are 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 has shown there are relatively slow variations in the measured parameters of volume and temperature.

SR 3.1.7.4 and SR 3.1.7.6

SR 3.1.7.4 verifies the continuity of the explosive charges in the injection valves to ensure that proper operation will occur if required. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day Frequency is based on operating experience and has demonstrated the reliability of the explosive charge continuity.

SR 3.1.7.6 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

(continued)

BASES

SURVEILLANCE REQUIREMENT

SR 3.1.7.4 and SR 3.1.7.6 (continued)

3

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.

sodium penta borate 3

SR 3.1.7.5

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. 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. 3

3 sodium pentaborate 3 1275 3

SR 3.1.7.7

40

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.7 (continued)

requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests 4 ~~inspections~~ confirm component OPERABILITY, ~~trend performance~~ 5, and detect incipient failures by indicating abnormal performance. The Frequency of this Surveillance is ~~(~~in accordance with the Inservice Testing Program ~~or 92 days)~~ 6.

SR 3.1.7.8 and SR 3.1.7.9

These Surveillances ensure that there is a functioning flow path from the boron solution storage tank to the RPV, including the firing of an explosive valve. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. The pump and explosive valve tested should be alternated such that both complete flow paths are tested every ~~36~~ 48 months at alternating ~~18~~ 24 month intervals. 3 The Surveillance may be performed in separate steps to prevent injecting boron into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a test tank through one SLC subsystem and into the RPV. The ~~18~~ 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the ~~18~~ 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Demonstrating that all heat traced piping between the boron solution storage tank and the suction inlet to the injection pumps is unblocked ensures that there is a functioning flow path for injecting the sodium pentaborate solution. An acceptable method for verifying that the suction piping is unblocked is to pump from the storage tank to the ~~test~~ storage tank. 1

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.8 and SR 3.1.7.9 (continued)

The ~~18~~ 24 month Frequency is acceptable since there is a low probability that the subject piping will be blocked due to precipitation of the boron from solution in the heat traced piping. This is especially true in light of the temperature verification of this piping required by SR 3.1.7.3. However, if, in performing SR 3.1.7.3, it is determined that the temperature of this piping has fallen below the specified minimum, SR 3.1.7.9 must be performed once within 24 hours after the piping temperature is restored to within the limits of Figure 3.1.7-2. 3

3

~~SR 3.1.7.10~~

~~Enriched sodium pentaborate solution is made by mixing granular, enriched sodium pentaborate with water. Isotopic tests on the granular sodium pentaborate to verify the actual B-10 enrichment must be performed prior to addition to the SLC tank in order to ensure that the proper B-10 atom percentage is being used.~~

REFERENCES

1. 10 CFR 50.62.
2. u FSAR, Section ~~4.2.3/4.3~~ 9.3.5.3. 6 11

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.1.7 - STANDBY LIQUID CONTROL SYSTEM

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

2. The proper 10 CFR 50.36(c)(2)(ii) criterion has been used. The current wording was developed prior to the issuance of the change to 10 CFR 50.36, which uses criterion 4 for the current words in the NUREG.

3. Changes have been made to reflect those changes made to the Specification. The following requirements have been renumbered, where applicable, to reflect these changes.

4. Typographical/grammatical error corrected.

5. The IST program for Quad Cities 1 and 2 is not required to provide information for trend purposes.

6. The brackets have been removed and the proper plant specific information/value has been provided.

8. Editorial change made for enhanced clarity.

B 3.1 REACTIVITY CONTROL SYSTEMS

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

BASES

BACKGROUND

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

1

APPLICABLE SAFETY ANALYSES

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

a. Close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within the limits of 10 CFR 100 (Ref. 2); and

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

Isolation of the SDV can also be accomplished by manual closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. For a bounding leakage case, the offsite doses are well within the limits of 10 CFR 100 (Ref. 2), and adequate core cooling is maintained (Ref. 3). The SDV vent and drain valves allow continuous drainage of the SDV during normal plant operation

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

to ensure that the SDV has sufficient capacity to contain the reactor coolant discharge during a full core scram. To automatically ensure this capacity, a reactor scram (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation") is initiated if the SDV water level in the instrument volume exceeds a specified setpoint. The setpoint is chosen so that all control rods are inserted before the SDV has insufficient volume to accept a full scram.

SDV vent and drain valves satisfy Criterion 3 of ~~the NRC Policy Statement~~. 10 CFR 50.36 (c)(2)(ii) 1

LCO

The OPERABILITY of all SDV vent and drain valves ensures that the SDV vent and drain valves will close during a scram to contain reactor water discharged to the SDV piping. Since the vent and drain lines are provided with two valves in series, the single failure of one valve in the open position will not impair the isolation function of the system. Additionally, the valves are required to open on scram reset to ensure that a path is available for the SDV piping to drain freely at other times. 2

APPLICABILITY

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

ACTIONS

The ACTIONS ~~t~~able is modified by a Note indicating that a separate Condition entry is allowed for each SDV vent and drain line. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions 3

(continued)

BASES

ACTIONS (continued)

for each inoperable SDV line. Complying with the Required Actions may allow for continued operation, and subsequent inoperable SDV lines are governed by subsequent Condition entry and application of associated Required Actions.

Insert Actions 4

A.1

When one SDV vent or drain valve is inoperable in one or more lines, ~~the valves must be restored to OPERABLE status within 7 days.~~ the line must be isolated to contain the reactor coolant during a scram. 4 The 7 day Completion Time is reasonable, given the level of redundancy in the lines and the low probability of a scram occurring while the valve(s) are inoperable and the line(s) not isolated 4. The SDV is still isolable since the redundant valve in the affected line is OPERABLE. During these periods, the single failure criterion may not be preserved, and a higher risk exists to allow reactor water out of the primary system during a scram.

B.1

If both valves in a line are inoperable, the line must be isolated to contain the reactor coolant during a scram. When a line is isolated, the potential for an inadvertent scram due to high SDV level is increased. ~~Required Action B.1 is modified by a Note that allows periodic draining and venting of the SDV when a line is isolated.~~ During these periods, the line may be unisolated under administrative control. This allows any accumulated water in the line to be drained, to preclude a reactor scram on SDV high level. This is acceptable since the administrative controls ensure the valve can be closed quickly, by a dedicated operator at the valve controls, if a scram occurs with the valve open. 4

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

C.1

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO

(continued)

4

Insert ACTIONS

The ACTIONS Table is modified by a second Note stating that an isolated line may be unisolated under administrative control to allow draining and venting of the SDV.

BASES

ACTIONS

C.1 (continued)

2

does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

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

1

Improper valve position (closed) would not affect the isolation function

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

SR 3.1.8.2

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

SR 3.1.8.3

SR 3.1.8.3 is an integrated test of the SDV vent and drain valves to verify total system performance. After receipt of a simulated or actual scram signal, the closure of the SDV vent and drain valves is verified. The closure time of

(continued)

BASES

SURVEILLANCE REQUIREMENTS

4

SR 3.1.8.3 (continued)

30 ~~60~~ seconds after receipt of a scram signal is based on the bounding leakage case evaluated in the accident analysis (Ref. 3 ~~2~~). Similarly, after receipt of a simulated or actual scram reset signal, the opening of the SDV vent and drain valves is verified. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1 and the scram time testing of control rods in LCO 3.1.3, "Control Rod OPERABILITY," overlap this Surveillance to provide complete testing of the assumed safety function. The 24 ~~18~~ month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 ~~18~~ month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

1 2 4 1

REFERENCES

1. 4 FSAR, Section [~~4.2.3.2.2.3~~ 4.6.3.3.2.8].

2. 10 CFR 100.

3. NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," August 1981.

5

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.1.8 - SDV VENT AND DRAIN VALVES

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

2. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

3. Typographical/grammatical error corrected.

4. Changes have been made to reflect those changes made to the Specification. The following requirements have been renumbered, where applicable, to reflect these changes.

5. The brackets have been removed and the proper plant specific information/value has been provided.

ADMINISTRATIVE CHANGES
("A.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change involves reformatting, renumbering, and rewording the existing Technical Specifications. The reformatting, renumbering, and rewording process involves no technical changes to the existing Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analyses assumptions. This change is administrative in nature. Therefore, the change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - MORE RESTRICTIVE
("M.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change provides more stringent requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change 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 the methods governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The imposition of more restrictive requirements either has no impact on or increases the margin of plant safety. As provided in the discussion of the change, each change in this category is by definition, providing additional restrictions to enhance plant safety. The change maintains requirements within the safety analyses and licensing basis. Therefore, this change does not involve a significant reduction in a margin of safety.

"GENERIC" LESS RESTRICTIVE CHANGES:
RELOCATING DETAILS TO TECHNICAL SPECIFICATION BASES, UFSAR, TRM, OR OTHER PLANT CONTROLLED DOCUMENTS
("LA.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change relocates certain details from the Technical Specifications to the Bases, UFSAR, TRM, or other plant controlled documents. The Bases, UFSAR, TRM, and other plant controlled documents containing the relocated information will be maintained in accordance with 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specification Bases are subject to the change control provisions in the Administrative Controls Chapter of the ITS. The UFSAR is subject to the change control provisions of 10 CFR 50.71(e), and the plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Since any changes to the Bases, UFSAR, TRM, or other plant controlled documents will be evaluated per the requirements of the Bases Control Program in Chapter 5.0 of the ITS or 10 CFR 50.59, no increase (significant or insignificant) in the probability or consequences of an accident previously evaluated will be allowed. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change 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 a change in the methods governing normal plant operation. The proposed change will not impose or eliminate any requirements, and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the details to be transposed from the Technical Specifications to the Bases, UFSAR, TRM, or other plant controlled

"GENERIC" LESS RESTRICTIVE CHANGES:
RELOCATING DETAILS TO TECHNICAL SPECIFICATION BASES, UFSAR, TRM, OR OTHER PLANT CONTROLLED DOCUMENTS
("LA.x" Labeled Comments/Discussions)

3. (continued)

documents are the same as the existing Technical Specifications. Since any future changes to these details in the Bases, UFSAR, TRM, or other plant controlled documents will be evaluated per the requirements of 10 CFR 50.59, no reduction (significant or insignificant) in a margin of safety will be allowed. Based on 10 CFR 50.92, the existing requirement for NRC review and approval of revisions, to these details proposed for relocation, does not have a specific margin of safety upon which to evaluate. However, since the proposed change is consistent with the BWR ISTS, NUREG-1433, Rev. 1, approved by the NRC Staff, revising the Technical Specifications to reflect the approved level of detail ensures no significant reduction in the margin of safety.

"GENERIC" LESS RESTRICTIVE CHANGES:
EXTENDING SURVEILLANCE FREQUENCIES FROM 18 MONTHS TO 24 MONTHS FOR SURVEILLANCES OTHER THAN CHANNEL CALIBRATIONS
("LD.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change involves a change in the surveillance testing intervals from 18 months to 24 months. The proposed change does not physically impact the plant nor does it impact any design or functional requirements of the associated systems. That is, the proposed change does not degrade the performance or increase the challenges of any safety systems assumed to function in the accident analysis. The proposed change does not impact the Surveillance Requirements themselves nor the way in which the Surveillances are performed. Additionally, the proposed change does not introduce any new accident initiators since no accidents previously evaluated have as their initiators anything related to the frequency of surveillance testing. The proposed change does not affect the availability of equipment or systems required to mitigate the consequences of an accident because of the availability of redundant systems or equipment and because other test performed more frequently will identify potential equipment problems. Furthermore, an historical review of surveillance test results indicated that all failures identified were unique, non-repetitive, and not related to any time-based failure modes, and indicated no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed change does not increase the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change involves a change in the surveillance testing intervals from 18 months to 24 months. The proposed change does not introduce any failure mechanisms of a different type than those previously evaluated since there are no physical changes being made to the facility. In addition, the Surveillance Requirements themselves and the way Surveillances are performed will remain unchanged. Furthermore, an historical review of surveillance test results indicated no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

"GENERIC" LESS RESTRICTIVE CHANGES:
EXTENDING SURVEILLANCE FREQUENCIES FROM 18 MONTHS TO 24 MONTHS FOR SURVEILLANCES OTHER THAN CHANNEL CALIBRATIONS
("LD.x" Labeled Comments/Discussions) (continued)

3. Does this change involve a significant reduction in a margin of safety?

Although the proposed change will result in an increase in the interval between surveillance tests, the impact on system availability is minimal based on other, more frequent testing or redundant systems or equipment, and there is no evidence of any failures that would impact the availability of the systems. Therefore, the assumptions in the licensing basis are not impacted, and the proposed change does not involve a significant reduction in a margin of safety.

RELOCATED SPECIFICATIONS
("R.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change relocates requirements and surveillances for structures, systems, components or variables that do not meet the criteria for inclusion in Technical Specifications as identified in the Application of Selection Criteria to the Quad Cities 1 and 2 Technical Specifications. The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected structures, systems, components or variables will be relocated from the Technical Specifications to an appropriate administratively controlled document which will be maintained pursuant to 10 CFR 50.59. In addition, the affected structures, systems, components or variables are addressed in existing surveillance procedures which are also controlled by 10 CFR 50.59 and subject to the change control provisions imposed by plant administrative procedures, which endorse applicable regulations and standards. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change 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 a change in the methods governing normal plant operation. The proposed change will not impose or eliminate any requirements and adequate control of existing requirements will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structure, system, component or variable remain the same as the existing Technical Specifications. Since any future changes to these requirements or the surveillance procedures will be evaluated per the requirements of 10 CFR 50.59, no reduction in a margin of safety will be permitted.

RELOCATED SPECIFICATIONS
("R.x" Labeled Comments/Discussions)

3. (continued)

The existing requirement for NRC review and approval of revisions, in accordance with 10 CFR 50.92, to these details proposed for relocation does not have a specific margin of safety upon which to evaluate. However, since the proposed change is consistent with the BWR ISTS, NUREG-1433, Rev. 1, approved by the NRC Staff, revising the Technical Specifications to reflect the approved level of detail ensures no significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.1.1 - SHUTDOWN MARGIN

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change allows control rods to be inserted at all times whenever SDM is not met in MODE 5. The insertion of control rods is not considered an initiator of any accidents previously evaluated, and therefore, will not affect their probability. The proposed change will also allow actions to remove fuel bundles, which could result in a fuel handling accident. However, the fuel handling accident assumes a bundle is dropped, and this change does not increase the probability of a dropped bundle. Additionally, the proposed actions allow negative reactivity additions to control the event and reduce the consequences. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change 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 compensatory actions to reduce the consequences of a previously analyzed event. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change allows operations to add negative reactivity when SDM is below the expected levels and results in a more expeditious correction of the required SDM. Therefore, the proposed change does not allow operations which would involve a significant reduction in the margin of safety.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change deletes the requirement for inserting insertable control rods in core cells with no fuel bundles. Normal control rod movement is not considered an initiator of a previously evaluated accident. Therefore, revising actions associated with control rod movement will not significantly increase the probability of an accident previously evaluated. Furthermore, since the reactivity effect of a control rod in a core cell with no fuel bundles is negligible, the lack of this insertion requirement will not involve a significant increase in the consequences of an accident previously evaluated.

2. Does the change 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. Movement of a control rod with no fuel assemblies in the core cell does not significantly affect the core reactivity, and therefore, does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   Considering that the negative reactivity inserted by removing the adjacent four fuel assemblies is significantly more than any minimal positive reactivity inserted during any movement of the associated control rod, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.1.2 - REACTIVITY ANOMALIES

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change would increase the ACTION time allowed to evaluate and determine the cause of any reactivity anomaly to 72 hours. 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 which predict 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 change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change 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. Does this change involve a significant reduction in a margin of safety?

   The proposed change would allow additional time to determine the cause of any reactivity anomaly 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 requiring an immediate shutdown which may cause a core transient while in this condition. Therefore, the proposed change does not allow operations which would involve a significant reduction in the margin of safety.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change revises the activities that result in a requirement to perform the Surveillance. The proposed activities are those that could have significantly altered the core reactivity and are not readily reversible. Those activities which alter core reactivity on a frequent basis as part of the normal operation, such as control rod movement are excluded. The performance of this Surveillance does not involve the operation of, or change to, any equipment which is assumed as an initiator for any analyzed accidents. The excluded operations are previously approved normal activities with reversible effects, which do not impact the consequences of any analyzed accidents. Therefore, this change will not significantly increase the probability or the consequences of an accident previously evaluated.

2. Does the change 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 does not introduce a new mode of operation. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change revises the activities that result in a requirement to perform the Surveillance. Not requiring this Surveillance to be performed following CORE ALTERATIONS which do not significantly affect the core reactivity does not impact the ability to maintain the plant within acceptable limits. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

L.3 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change revises the Surveillance Frequency for the verification of the reactivity difference between the monitored k_{eff} and the predicted k_{eff}. The proposed change continues to provide assurance that plant operation is maintained within the assumptions of the DBA and transient analysis. The proposed change in Frequency does not involve the operation of, or change to, any equipment assumed to be an initiator for any analyzed accidents. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

2. Does the change 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 does not introduce a new mode of operation. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The extension in the surveillance test interval is insignificant given that the proposed Frequency considers the relatively slow change in core reactivity with exposure, and operating experience related to variations in core reactivity. The proposed change does not impact the ability of the equipment to maintain the plant within acceptable limits. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.1.3 - CONTROL ROD OPERABILITY

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change revises the required ACTIONS for the local distribution of inoperable control rods to be applicable only when; 1) RTP is below 10%; and 2) the rods are in noncompliance with the analyzed rod position sequence. Additionally, 4 hours is proposed to be allowed for restoration. The applicability of actions associated with and the time periods allowed for restoration of inoperable rods are not assumed in the initiation of any accidents previously evaluated, and therefore, cannot increase the probability of such accidents. The current analyses place no restrictions on the local distribution of inoperable control rods for the excluded conditions. Therefore, this change does not contribute to an increase in the consequences of previously evaluated accidents. Additionally, the extended time for Action does not affect the ability of the system to respond to such accidents and also does not contribute to an increase in the consequences of an accident previously evaluated.

2. Does the change 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 changes, but provides additional time to complete the previously approved actions. Furthermore, this change eliminates some required actions for conditions which are allowed in the current analysis. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change allows additional time to correct control rod patterns which may not be analyzed. However, these conditions occur infrequently. Any safety impact as a result of the additional time is offset by allowing sufficient time to perform the required activities without undue haste. The safety benefit results from minimizing the potential for error and the plant transient associated with a forced shutdown if the activities are not completed in the required time. The other changes reflect operational allowances that are consistent with assumptions in safety analyses. Therefore, the proposed changes do not allow operations which would involve a significant reduction in the margin of safety.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change revises the time allowed to disarm an inoperable stuck control rod. The time period allowed to disarm inoperable rods is not assumed in the initiation of any accidents previously evaluated, and therefore, cannot increase the probability of such accidents. Additionally, since this change does not affect the actual control rod position, and the analysis is insensitive to one inoperable fully withdrawn control rod, the extended time for action does not affect the ability of the system to respond to accidents. Therefore, this change does not contribute to an increase in the consequences of an accident previously evaluated.

2. Does the change 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 does not introduce a new mode of operation. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change allows additional time to disarm an inoperable stuck control rod. However, the control rod is in a position allowed by the safety analysis; disarming only deters future misoperation and potential damage. Such misoperation is of low probability during the time immediately following the original discovery of the inoperable control rod. Any safety impact as a result of the additional time is offset by allowing sufficient time to perform the required activities without undue haste. The safety benefit results from minimizing the potential for error and the plant transient associated with a forced shutdown if the activities are not completed in the required time. Therefore, the proposed change does not allow operations which would involve a significant reduction in the margin of safety.

L.3 CHANGE

Not used.

L.4 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change provides an extended time to perform a SDM Surveillance after identifying a stuck rod. A single control rod stuck in a withdrawn position does not affect the capability of the remaining OPERABLE control rods to provide the required scram and shutdown reactivity. Therefore, this extended time frame to perform the Surveillance will not significantly increase the probability of an accident previously evaluated. Furthermore, since the remaining OPERABLE control rods provide the required scram and shutdown reactivity, this change will not involve a significant increase in the consequences of an accident previously evaluated.

2. Does the change 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 and does not introduce a new mode of operation. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   A notch test is promptly performed (within 24 hours) for each of the remaining withdrawn control rods to ensure no additional control rods are stuck. With this assurance the extension of the time allowed to demonstrate SHUTDOWN MARGIN provides a reasonable time to confirm that the SDM is still maintained. This result is expected because prior analysis includes sufficient uncertainties and biases to account for the stuck rod. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

L.5 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change revises the time allowed to fully insert and disarm an inoperable control rod. The period allowed to fully insert and disarm inoperable rods is not assumed in the initiation of any accidents previously evaluated and therefore cannot increase the probability of such accidents. Additionally, the extended time for action does not affect the ability of the system to respond to such accidents, since a single control rod is assumed to be withdrawn in the accident analyses. Therefore, this change does not involve a significant increase in the consequences of an accident previously evaluated.

2. Does the change 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 does not introduce a new mode of operation. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change allows additional time to insert and disarm inoperable control rods. However, the control rod is assumed to be fully withdrawn in the accident analysis. Any safety impact as a result of this additional time is offset by allowing sufficient time to perform the required activities without undue haste. The safety benefit results from minimizing the potential for error and the plant transient associated with a forced shutdown if the activities are not completed in the required time. Therefore, the proposed change does not allow operations which would involve a significant reduction in the margin of safety.

L.6 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change deletes the requirement for increased frequency of control rod testing when more than three rods exceed the maximum scram time. The Frequency of scram time testing control rods is not assumed in the initiation of any accidents previously evaluated and therefore cannot increase the probability of such accidents. Additionally, the current analysis provides sufficient margin to account for the proposed allowances of slow and inoperable control rods. Therefore, this change will not involve a significant increase in the consequences of an accident previously evaluated.

2. Does the change 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. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change is consistent with the assumptions of the current safety analysis and therefore does not involve a significant reduction in a margin of safety.

L.7 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change deletes the requirements for control rod coupling during the refueling mode. During refueling only one control rod is allowed to be withdrawn from core cells containing fuel assemblies. Therefore, the coupling requirements provide no required protection and the elimination does not increase the probability of a previously evaluated accident. Additionally, the remaining requirements provide controls consistent with the assumptions of the current analysis. Therefore, this change will not involve a significant increase in the consequences of an accident previously evaluated.

2. Does the change 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 and does not introduce a new mode of operation. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change removes uncredited controls and is consistent with the assumptions of the current safety analysis. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

L.8 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change would increase the time allowed to accomplish recoupling and allow bypassing of the RWM to recouple. These restrictions on recoupling control rods are not assumed in the initiation of any accidents previously evaluated. Therefore, changes to these restrictions cannot increase the probability of such accidents. Additionally, the proposed ACTION does not affect the ability of the systems to respond to such accidents since a number of inoperable control rods are assumed in the accident analyses. Therefore, the change does not contribute to an increase in the consequences of an accident previously evaluated.

2. Does the change 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 does not introduce an new mode of operation. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change removes unnecessary restrictions which may prevent an unnecessary shutdown and is consistent with the assumptions of the current safety analysis. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

L.9 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The requirement to demonstrate the affected control rod does not go to the overtravel position to verify recoupling is not assumed in the initiation of any analyzed event. This requirement was specified in the Technical Specifications to ensure recoupling was positively verified. The proposed deletion of this explicit requirement is considered administrative since SR 3.0.1 requires the appropriate SRs to be performed to demonstrate OPERABILITY after restoration of a component that caused the SR to be failed. In this case, SR 3.0.1 would require SR 3.1.3.5 to be performed which verifies the affected control rod does not go to the withdrawn overtravel position. As a result, the accident consequences are unaffected by this change. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change 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 modification to the plant.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed deletion of the explicit requirement to demonstrate a recoupled control rod does not go to the withdrawn overtravel position is considered administrative since SR 3.0.1 requires the appropriate SRs to be performed to demonstrate OPERABILITY after restoration of a component that caused the SR to be failed. In this case, SR 3.0.1 would require SR 3.1.3.5 to be performed which verifies the affected control rod does not go to the withdrawn overtravel position. As a result, the existing requirement to verify control rod coupling integrity after recoupling of the affected control rod is maintained. Therefore, this change does not involve a significant reduction in a margin of safety.

L.10 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The method used to verify control rod coupling is not assumed to be an initiator of any analyzed event. The change continues to require control rod coupling to be verified. SR 3.1.3.5 requires all fully withdrawn rods be subjected to verification of coupling by the overtravel test. SR 3.1.3.5 also requires the overtravel test to be performed prior to declaring a control rod OPERABLE after work on a control rod or CRD System that could affect coupling. As a result, the consequences of an event occurring due to a control rod being uncoupled are unaffected by this change. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change 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 modification to the plant.

3. Does this change involve a significant reduction in a margin of safety?

   A margin of safety is not reduced. A response to control rod motion on nuclear instrumentation is indicative that a control rod is following its drive but gives no indication as to whether a control rod is coupled. Likewise, failure to have a response to rod motion on nuclear instrumentation does not indicate that a rod is uncoupled. Although operators will continue to monitor nuclear instrumentation response during control rod motion, the results are insufficiently conclusive to use the results as a surveillance test for the verification of rod coupling. SR 3.1.3.5 requires all fully withdrawn rods be subjected to verification of coupling by the overtravel test. The overtravel test provides a positive check of coupling integrity since only an uncoupled control rod can go to the overtravel position. SR 3.1.3.5 also requires the overtravel test to be performed prior to declaring a control rod OPERABLE after work on a control rod or CRD System than could affect coupling. Therefore, control rod coupling integrity is still adequately verified and this change does not involve a significant reduction in a margin of safety.

L.11 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The control rod position indication system is not assumed in the initiation of any analyzed event. The requirement to determine the control rod position indication system is OPERABLE by the performance of the control rod movement verification surveillance does not need to be explicitly stated in the Technical Specifications. To perform control rod movement tests required by SR 3.1.3.2 and SR 3.1.3.3, position indication must be available. If position indication is not available, these tests cannot be satisfied and the appropriate actions must be taken for inoperable control rods in accordance with the ACTIONS of ITS 3.1.3. As a result, accident consequences are unaffected by this change. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change 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 modification to the plant.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed deletion of the requirement to determine the control rod position indication system is OPERABLE by the performance of the control rod movement verification surveillance does not impact any margin of safety. To perform control rod movement tests required by SR 3.1.3.2 and SR 3.1.3.3, position indication must be available. If position indication is not OPERABLE, these tests cannot be satisfied and the appropriate actions must be taken for inoperable control rods in accordance with the ACTIONS of ITS 3.1.3. As a result, position indication will be maintained OPERABLE to satisfy the associated SRs of Specification 3.1.3 without the need for explicit position indication requirements in the Technical Specifications. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.1.4 - CONTROL ROD SCRAM TIMES

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change modifies the Surveillance Frequency for scram time testing of all control rods. The change does not affect equipment design or operation. The affected Surveillance is not considered to be an initiator of any analyzed event. Therefore, this change does not significantly increase the probability of a previously analyzed accident. This change only requires control rod scram time testing for a control rod following fuel movement in the associated core cell instead of testing all of the control rods following CORE ALTERATIONS. This change is acceptable since the intent of testing all of the control rods following CORE ALTERATIONS ensures the overall negative reactivity insertion rate is maintained following refueling activities that may impact a significant number of control rods (e.g., CRD replacement, CRDM overhaul, or movement of fuel in the core cell). When only a few control rods have been impacted by fuel movement, the effect on the overall negative reactivity insertion rate is insignificant. Scram time testing will still be required for the control rod(s) affected by any fuel movement. It is expected that during a refueling outage, all control rods will be affected. Therefore, this change does not impact safety analysis assumptions and does not involve a significant increase in the consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The modification of the Surveillance Frequency does not involve physical modification to the plant and does not introduce a new mode of operation. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The change in the Surveillance Frequency only requires scram time testing of those control rods affected by fuel movement. The impact, as a result of this change, on the negative reactivity insertion rate is insignificant since certain fuel movements may only impact a small percentage of control rods. In this condition, the proposed change requires scram time testing of the affected control rods. Scram time testing of all control rods is still required following a refueling outage where the negative reactivity

L.1 CHANGE

3. (continued)

insertion rate of a large number of control rods could have been impacted since it is expected that all control rods will be affected. In addition, this change is considered acceptable since the most common outcome of the performance of a Surveillance is the successful demonstration that the acceptance criteria are satisfied. This change reduces the amount of control rod testing, thereby, increasing control rod reliability. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.1.5 - CONTROL ROD SCRAM ACCUMULATORS

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change 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 to make the declaration. Inoperable accumulators are not considered initiators for any accidents previously evaluated, and therefore, cannot increase the probability of such accidents. Additionally, the current analysis provides sufficient margin to account for the proposed allowances of slow and inoperable control rods. Therefore, this change will not involve a significant increase in the consequences of an accident previously evaluated.

2. Does the change 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. The change in the operation is consistent with current safety analysis assumptions. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change 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 and/or charging water header 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.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change would allow a short time to attempt to return inoperable accumulators to service 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 corrected. Therefore, the change does not involve a significant increase in the probability of an accident previously evaluated. Additionally, the proposed actions are the same as the current actions except for the additional time allowed, therefore the actions have been previously considered and this change will not involve a significant increase in the consequence of an accident previously evaluated.

2. Does the change 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. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change is consistent with the assumptions of the current safety analysis and provides for consistent actions, but allows sufficient time to restore OPERABILITY and prevent a transient. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.1.6 - ROD PATTERN CONTROL

There were no plant specific less restrictive changes identified for this Specification.

There were no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.1.8 - SDV VENT AND DRAIN VALVES

There were no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
CTS: 3/4.3.J - CONTROL ROD DRIVE HOUSING SUPPORT

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The CRD housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a CRD housing failure. The CRD housing support is not an accident initiator or precursor and, as such, cannot contribute to an increase in the probability of an accident previously evaluated. The relocation of this Specification does not result in the removal of the requirement to verify proper installation of the CRD housing support. Plant configuration management controls ensure through post-maintenance testing and inspections that the proper configuration for the CRD housing supports is maintained. These controls are currently in place and are used to ensure this system and other plant systems are properly configured prior to being considered OPERABLE for plant operation. Based on the controls that the plant has in place to ensure the CRD housing support is properly installed, the change does not involve a significant increase in the consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve physical alteration of the plant (no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. The proposed change does not impose requirements different from those being used for normal post-maintenance inspections to ensure the CRD housing support is properly installed. The proposed change will rely on plant configuration management controls to ensure that this system and other plant systems are returned to their design configuration condition. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

L.1 CHANGE (continued)

3. Does this change involve a significant reduction in a margin of safety?

The CRD housing support Technical Specification ensures proper installation of this system during MODES 1, 2, and 3. The installation checks are performed while the plant is shutdown and are necessary only after work has been done to alter the system configuration. These post-maintenance checks are currently performed by procedural control on this and other plant systems. The use of present plant configuration management controls will ensure that these systems meet design requirements. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
CTS: 3/4.3.N - ECONOMIC GENERATION CONTROL SYSTEM

There were no plant specific less restrictive changes identified for this Specification.

ENVIRONMENTAL ASSESSMENT
ITS: SECTION 3.1 - REACTIVITY CONTROL SYSTEMS

In accordance with the criteria set forth in 10 CFR 50.21, ComEd has evaluated this proposed Technical Specification change for identification of licensing and regulatory actions requiring environmental assessment, determined it meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9) and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92(b). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or which changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria:

1. The amendment involves no significant hazards consideration.

   As demonstrated in the No Significant Hazards Consideration, this proposed amendment does not involve any significant hazards consideration.

2. There is no significant change in the type or significant increase in the amounts of any effluents that may be released offsite.

   The proposed change will not result in changes in the operation or configuration of the facility. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no change in the types or significant increase in the amounts of any effluents released offsite resulting from this change.

3. There is no significant increase in individual or cumulative occupational radiation exposure.

   The proposed change will not result in changes in the operation or configuration of the facility which impact radiation exposure. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from this change.

Therefore, based upon the above evaluation, ComEd has concluded that no irreversible consequences exist with the proposed change.

3.2 POWER DISTRIBUTION LIMITS

3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

LCO 3.2.1 All APLHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER $\geq$ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any APLHGR not within limits.	A.1 Restore APLHGR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.1.1 Verify all APLHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after $\geq$ 25% RTP <u>AND</u> 24 hours thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

LCO 3.2.2 All MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Any MCPR not within limits.	A.1	Restore MCPR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify all MCPRs are greater than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.2.2 Determine the MCPR limits.	Once within 72 hours after each completion of SR 3.1.4.1 AND Once within 72 hours after each completion of SR 3.1.4.2 AND Once within 72 hours after each completion of SR 3.1.4.4

3.2 POWER DISTRIBUTION LIMITS

3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

LCO 3.2.3 All LHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER $\geq$ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any LHGR not within limits.	A.1 Restore LHGR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.3.1 Verify all LHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after $\geq$ 25% RTP AND 24 hours thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.4 Average Power Range Monitor (APRM) Gain and Setpoint

LCO 3.2.4

a. FDLRC and the ratio of MFLPD to Fraction of RTP (FRTP) shall be less than or equal to 1.0; or

b. Each required APRM Flow Biased Neutron Flux — High Function Allowable Value shall be modified by the lesser of 1/FDLRC or FRTP/MFLPD; or

c. Each required APRM gain shall be adjusted such that the APRM readings are ≥ 100% times the higher of FRTP times FDLRC or of MFLPD.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	6 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.4.1	-------------------NOTE------------------- Not required to be met if SR 3.2.4.2 is satisfied for LCO 3.2.4.b or LCO 3.2.4.c requirements. -- Verify FDLRC and the ratio of MFLPD to FRTP are within limits.	Once within 12 hours after $\geq$ 25% RTP <u>AND</u> 24 hours thereafter
SR 3.2.4.2	-------------------NOTE------------------- Not required to be met if SR 3.2.4.1 is satisfied for LCO 3.2.4.a requirements. -- Verify each required: a. APRM Flow Biased Neutron Flux — High Function Allowable Value is modified by less than or equal to the lesser of 1/FDLRC or FRTP/MFLPD; or b. APRM gain is adjusted such that the APRM reading is $\geq$ 100% times the higher of FRTP times FDLRC or of MFLPD.	12 hours

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that the criteria specified in 10 CFR 50.46 are met during the postulated design basis loss of coolant accident (LOCA). Additionally, for General Electric fuel types, APLHGR limits are specified to ensure that the fuel design limits identified in Reference 1 are not exceeded during anticipated operational occurrences (AOOs).

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs), that determine the APLHGR limits are presented in References 1, 2, 3, 4, and 5.

LOCA analyses are performed to ensure that the determined APLHGR limits are adequate to meet the peak cladding temperature (PCT) and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using calculational models that are consistent with the requirements of 10 CFR 50, Appendix K. A complete discussion of the analysis code is provided in References 1 and 5. The PCT following a postulated LOCA is a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within an assembly. A conservative multiplier is applied to the LHGR and APLHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR. For GE fuel, the APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by the minimum anticipated local peaking factor. For Siemens Power Corporation fuel, APLHGR limits are typically set high enough such that the LHGR limits are more limiting than the APLHGR limits.

For single recirculation loop operation, a conservative multiplier is applied to the exposure dependent APLHGR limits for two loop operation (Ref. 6). This additional limitation is due to the conservative analysis assumption of

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	an earlier departure from nucleate boiling with one recirculation loop available, resulting in a more severe cladding heatup during a LOCA. For GE fuel types, the APLHGR limits also incorporate the results of the fuel design limits. The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 1, 2, 3 and 4. Fuel design evaluations are performed to demonstrate that the 1% limit on the fuel cladding plastic strain and other fuel design limits described in Reference 1 are not exceeded during AOOs for operation with LHGRs up to the operating limit LHGR. APLHGR limits are equivalent to the LHGR limit for each fuel rod divided by the local peaking factor of the fuel assembly. APLHGR limits are developed as a function of exposure to ensure adherence to fuel design limits during the limiting AOOs (Ref. 4). The APLHGR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The APLHGR limits specified in the COLR are the result of the fuel design, DBA, and transient analyses. For two recirculation loops operating, the limit is dependent on exposure. With only one recirculation loop in operation, in conformance with the requirements of LCO 3.4.1, "Recirculation Loops Operating," the limit is determined by multiplying the exposure dependent APLHGR limit by a conservative multiplier determined by a specific single recirculation loop analysis (Ref. 6).
APPLICABILITY	The APLHGR limits are primarily derived from fuel design evaluations and LOCA and transient analyses that are assumed to occur at high power levels. Studies and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels ≤ 25% RTP, the reactor is operating with substantial margin to the APLHGR limits; thus, this LCO is not required.

(continued)

BASES (continued)

ACTIONS

A.1

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

B.1

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

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

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

(continued)

BASES (continued)

REFERENCES

1. NEDE-24011-P-A "General Electric Standard Application for Reactor Fuel" (as specified in Technical Specification 5.6.5).

2. UFSAR, Chapter 4.

3. UFSAR, Chapter 6.

4. UFSAR, Chapter 15.

5. EMF-94-217(NP), Revision 1, "Boiling Water Reactor Licensing Methodology Summary," November 1995.

6. UFSAR, Section 6.3.3.2.2.4.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR Safety Limit (SL) is set such that 99.9% of the fuel rods are expected to avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2). The operating limit MCPR is established to ensure that no fuel damage results during anticipated operational occurrences (AOOs). Although fuel damage does not necessarily occur if a fuel rod actually experienced boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

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

APPLICABLE SAFETY ANALYSES

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow state ($MCPR_f$) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency as identified in UFSAR, Chapter 15 (Ref. 5).

Flow dependent MCPR limits are determined by steady state thermal hydraulic methods with key physics response inputs benchmarked using the three dimensional BWR simulator code (Ref. 8) and a multichannel thermal hydraulic code (Ref. 9) to analyze slow flow runout transients on a cycle-specific basis. For core flows less than rated, the established MCPR operating limit is adjusted to provide protection of the MCPR SL in the event of an uncontrolled recirculation flow increase to the physical limit of the pump. Protection is provided for manual and automatic flow control by applying appropriate flow dependent MCPR operating limits. The MCPR operating limit for a given flow state is the greater of the rated conditions MCPR operating limit or the flow dependent MCPR operating limit. For automatic flow control, in addition to protecting the MCPR SL during the flow run-up event, protection is provided by the flow dependent MCPR operating limit to prevent exceeding the rated flow MCPR operating limit during an automatic flow increase to rated core flow. The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.

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

LCO

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

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a low recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting

(continued)

BASES

APPLICABILITY (continued)

transient occurs. Statistical analyses indicate that the nominal value of the initial MCPR expected at 25% RTP is > 3.5. Studies of the variation of limiting transient behavior have been performed over the range of power and flow conditions. These studies encompass the range of key actual plant parameter values important to typically limiting transients. The results of these studies demonstrate that a margin is expected between performance and the MCPR requirements, and that margins increase as power is reduced to 25% RTP. This trend is expected to continue to the 5% to 15% power range when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor provides rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < 25% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required.

ACTIONS

A.1

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

B.1

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

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

SR 3.2.2.2

Because the transient analyses take credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analyses. SR 3.2.2.2 determines the actual scram speed distribution and compares it with the assumed distribution. The MCPR operating limit is then determined based on either the applicable limit associated with the scram times of LCO 3.1.4, "Control Rod Scram Times," or the realistic scram times. The MCPR limit, including the scram insertion times for rated and off-rated flow conditions, are contained in the COLR. This determination must be performed once within 72 hours after each set of scram time tests required by SR 3.1.4.1, SR 3.1.4.2, and SR 3.1.4.4 because the effective scram speed distribution may change during the cycle or after maintenance that could affect scram times. The 72 hour Completion Time is acceptable due to the relatively minor changes in the actual scram speed distribution expected during the fuel cycle.

REFERENCES

1. NUREG-0562, June 1979.

2. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (as specified in Technical Specification 5.6.5).

(continued)

BASES

REFERENCES (continued)

3. UFSAR, Chapter 4.

4. UFSAR, Chapter 6.

5. UFSAR, Chapter 15.

6. EMF-94-217(NP), Revision 1, "Boiling Water Reactor Licensing Methodology Summary," November 1995.

7. NFSR-091, Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods, Commonwealth Edison Topical Report, (as specified in Technical Specification 5.6.5).

8. XN-NF-80-19(P)(A), Volume 1, Exxon Nuclear Methodology for Boiling Water Reactors - Neutronics Methods for Design and Analysis, (as specified in Technical Specification 5.6.5).

9. XN-NF-80-19(P)(A), Volume 3, Exxon Nuclear Methodology for Boiling Water Reactors - THERMEX Thermal Limits Methodology Summary Description, (as specified in Technical Specification 5.6.5).

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences (AOOs). Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure, or inability to cool the fuel does not occur during the normal operations and anticipated operating conditions identified in References 1 and 2.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design are presented in References 1 and 2. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR, Parts 20, 50, and 100. A mechanism that could cause fuel damage during normal operations and operational transients and that is considered in fuel evaluations is a rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet.

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

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limit specified in the COLR. The analysis also includes allowances for short term transient excursions above the operating limit while still remaining within the AOO limits, plus an allowance for densification power spiking.

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

(continued)

BASES (continued)

LCO

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

APPLICABILITY

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

ACTIONS

A.1

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

B.1

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS	SR 3.2.3.1 The LHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. They are compared to the LHGR limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slow changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at lower power levels.
REFERENCES	1. UFSAR, Chapter 4. 2. UFSAR, Chapter 15. 3. NUREG-0800, Section 4.2.II.A.2(g), Revision 2, July 1981.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 Average Power Range Monitor (APRM) Gain and Setpoint

BASES

BACKGROUND

The OPERABILITY of the APRMs and their setpoints is an initial condition of all safety analyses that assume rod insertion upon reactor scram. Applicable general design criteria are discussed in UFSAR, Sections 3.1.2.1, 3.1.3.2, 3.1.3.4, 3.1.3.5, and 3.1.4.8 (Ref. 1). This LCO is provided to require the APRM gain or APRM Flow Biased Neutron Flux—High Function Allowable Value (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b) to be adjusted when operating under conditions of excessive power peaking to maintain acceptable margin to the fuel cladding integrity Safety Limit (SL) and the fuel cladding 1% plastic strain limit.

For General Electric (GE) fuel, the condition of excessive power peaking is determined by the ratio of the actual power peaking to the limiting power peaking at RTP. This ratio is equal to the ratio of the core limiting MFLPD to the Fraction of RTP (FRTP), where FRTP is the measured THERMAL POWER divided by the RTP. Excessive power peaking exists when:

$$\frac{MFLPD}{FRTP} > 1,$$

indicating that MFLPD is not decreasing proportionately to the overall power reduction, or conversely, that power peaking is increasing. For Siemens (SPC) fuel, the condition of excessive power peaking is determined by Fuel Design Limit Ratio for Centerline Melt (FDLRC), which is defined as:

$$FDLRC = \frac{(LHGR)(1.2)}{(TLHGR)(FRTP)};$$

where LHGR is the Linear Heat Generation Rate, FRTP is the Fraction of Rated Thermal Power, and TLHGR is the Transient Linear Heat Generation Rate limit. The TLHGR limit is specified in the COLR and protects against fuel centerline melting and the fuel cladding 1% plastic strain during transient conditions throughout the life of the fuel.

(continued)

BASES

BACKGROUND (continued)	To maintain margins similar to those at RTP conditions, the excessive power peaking is compensated by a gain adjustment on the APRMs or modification of the APRM Neutron Flux - High Function Allowable Value. Either of these adjustments has effectively the same result as maintaining FDLRC and the ratio of MFLPD to FRTP less than or equal to 1.0 and thus maintains RTP margins for APLHGR, MCPR, and LHGR. Adjustments are based on the lowest APRM Neutron Flux - High Function Allowable Value or highest APRM reading resulting from the two methods (GE or Siemens). The normally selected APRM Flow Biased Neutron Flux - High Function Allowable Value positions the scram above the upper bound of the normal power/flow operating region that has been considered in the design of the fuel rods. The Allowable Value is flow biased with a slope that approximates the upper flow control line, such that an approximately constant margin is maintained between the flow biased trip level and the upper operating boundary for core flows in excess of about 45% of rated core flow. In the range of infrequent operations below 45% of rated core flow, the margin to scram is reduced because of the nonlinear core flow versus drive flow relationship. The normally selected APRM Allowable Value is supported by the analyses presented in Reference 2 that concentrate on events initiated from rated conditions. Design experience has shown that minimum deviations occur within expected margins to operating limits (APLHGR, MCPR, and LHGR), at rated conditions for normal power distributions. However, at other than rated conditions, control rod patterns can be established that significantly reduce the margin to thermal limits. Therefore, the APRM Flow Biased Neutron Flux - High Function Allowable Value may be reduced during operation when FDLRC or the combination of THERMAL POWER and MFLPD indicates an excessive power peaking distribution.
APPLICABLE SAFETY ANALYSES	The acceptance criteria for the APRM gain or setpoint adjustments are that acceptable margins (to APLHGR, MCPR, and LHGR) be maintained to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

UFSAR safety analyses (Ref. 2) concentrate on the rated power condition for which the minimum expected margin to the operating limits (APLHGR, MCPR, and LHGR) occurs. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limit the initial margins to these operating limits at rated conditions so that specified acceptable fuel design limits are met during transients initiated from rated conditions. At initial power levels less than rated levels, the margin degradation of the APLHGR, the MCPR, or the LHGR during a transient can be greater than at the rated condition event. This greater margin degradation during the transient is primarily offset by the larger initial margin to limits at the lower than rated power levels. However, power distributions can be hypothesized that would result in reduced margins to the pre-transient operating limit. When combined with the increased severity of certain transients at other than rated conditions, the fuel design limits and MCPR SL could be approached. At substantially reduced power levels, highly peaked power distributions could be obtained that could reduce thermal margins to the minimum levels required for transient events. To prevent or mitigate such situations, either the APRM gain is adjusted upward by the higher of the core limiting value of FDLRC or the ratio of the core limiting MFLPD to the FRTP, or the APRM Flow Biased Neutron Flux—High Function Allowable Value is required to be reduced by the lesser of either the reciprocal of the core limiting FDLRC or by the ratio of FRTP to the core limiting MFLPD. Either of these adjustments effectively counters the increased severity of some events at other than rated conditions by proportionally increasing the APRM gain or proportionally lowering the APRM Flow Biased Neutron Flux—High Function Allowable Value, dependent on the increased peaking that may be encountered.

The APRM gain and setpoint satisfy Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Meeting any one of the following conditions ensures acceptable operating margins for events described above:

a. Limiting excess power peaking;

(continued)

BASES

LCO (continued)

b. Reducing the APRM Flow Biased Neutron Flux—High Function Allowable Value by multiplying the APRM Flow Biased Neutron Flux—High Function Allowable Value by the lesser of either 1/FDLRC or the ratio of FRTP and the core limiting value of MFLPD; or

c. Increasing APRM gains to cause the APRM to read greater than or equal to 100 (%) times the higher of the core limiting value of FDLRC times FRTP or the core limiting MFLPD. This condition is to account for the reduction in margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

For GE fuel, MFLPD is the ratio of the limiting LHGR to the LHGR limit for the specific bundle type. For Siemens fuel, FDLRC times FRTP is the ratio of the LHGR times 1.2 to TLHGR. As power is reduced, if the design power distribution is maintained, MFLPD and FDLRC are reduced in proportion to the reduction in power. However, if power peaking increases above the design value, the MFLPD and FDLRC are not reduced in proportion to the reduction in power. Under these conditions, the APRM gain is adjusted upward or the APRM Flow Biased Neutron Flux—High Function Allowable Value is reduced accordingly. When the reactor is operating with peaking less than the design value, it is not necessary to modify the APRM Flow Biased Neutron Flux—High Function Allowable Value. Adjusting APRM gain or modifying the APRM Flow Biased Neutron Flux—High Function Allowable Value is equivalent to maintaining FDLRC and the ratio of MFLPD to FRTP less than or equal to 1.0, as stated in the LCO.

For compliance with LCO 3.2.4.b (APRM Flow Biased Neutron Flux—High Function Allowable Value modification) or LCO 3.2.4.c (APRM gain adjustment), only APRMs required to be OPERABLE per LCO 3.3.1.1, Function 2.b are required to be modified or adjusted. In addition, each APRM may be allowed to have its gain adjusted or Allowable Value modified independently of other APRMs that are having their gain adjusted or Allowable Value modified.

(continued)

BASES (continued)

APPLICABILITY

The FDLRC or the ratio of MFLPD to FRTP limit, APRM gain adjustment, or APRM Flow Biased Neutron Flux—High Function Allowable Value modification are provided to ensure that the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit are not violated during design basis transients. As discussed in the Bases for LCO 3.2.1, LCO 3.2.2, and LCO 3.2.3 sufficient margin to these limits exists below 25% RTP and, therefore, these requirements are only necessary when the reactor is operating at ≥ 25% RTP.

ACTIONS

A.1

If the APRM gain or Flow Biased Neutron Flux—High Function Allowable Value is not within limits while FDLRC or the ratio of MFLPD to FRTP exceed 1.0, the margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit may be reduced. Therefore, prompt action should be taken to restore FDLRC and the ratio of MFLPD to FRTP to within its required limit or make acceptable APRM adjustments such that the plant is operating within the assumed margin of the safety analyses.

The 6 hour Completion Time is normally sufficient to restore either FDLRC and the ratio of MFLPD to FRTP to within limits or to adjust the APRM gain or modify the APRM Flow Biased Neutron Flux—High Function Allowable Value to within limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LCO not met.

B.1

If FDLRC and the ratio of MFLPD to FRTP or the APRM Flow Biased Neutron Flux—High Function Allowable Value cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER is reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2

FDLRC and the ratio of MFLPD to FRTP is required to be calculated and compared to 1.0 or APRM gain adjusted or APRM Flow Biased Neutron Flux—High Function Allowable Value modified to ensure that the reactor is operating within the assumptions of the safety analysis. These SRs are only required to determine FDLRC and the ratio of MFLPD to FRTP and, assuming either exceeds 1.0, determine the appropriate APRM gain or APRM Flow Biased Neutron Flux—High Function Allowable Value and are not intended to be a CHANNEL FUNCTIONAL TEST for the APRM gain or Flow Biased Neutron Flux—High Function circuitry. SR 3.2.4.1 and SR 3.2.4.2 have been modified by Notes, which clarify that the respective SR does not have to be met if the alternate requirement demonstrated by the other SR is satisfied. The 24 hour Frequency of SR 3.2.4.1 is chosen to coincide with the determination of other thermal limits, specifically those for the APLHGR (LCO 3.2.1), MCPR (LCO 3.2.2), and LHGR (LCO 3.2.3). The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to APLHGR, MCPR, and LHGR operating limits at low power levels.

The 12 hour Frequency of SR 3.2.4.2 is required when either FDLRC or the ratio of MFLPD to FRTP is greater than 1.0, because more rapid changes in power distribution are typically expected.

REFERENCES

1. UFSAR, Sections 3.1.2.1, 3.1.3.2, 3.1.3.4, 3.1.3.5, and 3.1.3.8.
2. UFSAR, Chapter 15.

3.11 - LIMITING CONDITIONS FOR OPERATION

A. AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LCO 3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGR) shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: A.2

OPERATIONAL MODE 1 when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

ACTION A With an APLHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT:

1. ~~Initiate corrective ACTION within 15 minutes, and~~ LA.1
2. Restore APLHGR to within the required limit within 2 hours.

ACTION B With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

A. AVERAGE PLANAR LINEAR HEAT GENERATION RATE

SR 3.2.1.1 The APLHGRs shall be verified to be equal to or less than the limits specified in the CORE OPERATING LIMITS REPORT.

1. At least once per 24 hours,
2. Within 12 hours after ~~completion of a THERMAL POWER increase of at least 15%~~ ≥ 25% of RATED THERMAL POWER, and L.1
3. ~~Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.~~ L.2
4. ~~The provisions of Specification 4.0.D are not applicable.~~ L.1

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretation). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 The Applicability for CTS 3/4.11.A is "OPERATIONAL MODE 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER." With THERMAL POWER ≥ 25% RTP, the unit will always be in MODE 1. Therefore, it is unnecessary to state "OPERATIONAL MODE 1" in the Applicability of CTS 3/4.11.A (ITS 3.2.1).

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The requirement in CTS 3.11.A ACTION 1 to "Initiate corrective action within 15 minutes" to restore the limit is proposed to be relocated to the Bases in the form of a discussion that "prompt action" should be taken to restore the parameter to within the limits. Immediate action may not always be the conservative method to assure safety. The ITS 3.2.1 ACTION A 2 hour Completion Time for restoration allows appropriate actions to be evaluated by the operator and completed in a timely manner. As such, the relocated requirement is not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

"Specific"

L.1 CTS 4.11.A.2 is proposed to be changed to eliminate confusion as to how often the current Surveillance is required (e.g., after every 15% power change or at the end of any single power increase greater than 15%). Verifying the parameter within 12 hours of reaching or exceeding 25% RTP will generally require that the Surveillance be performed sooner than the CTS 4.11.A.2 requirement of "after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER," but would also reduce the number of times the Surveillance must be conducted during a startup if it is conducted after every 15% power change. A single verification is considered sufficient during initial startup considering the large inherent margin to operating limits at low power levels. Following the initial verification, the Surveillance is performed every 24 hours to identify any trends in these parameters that may lead to long term noncompliance. In addition, this change allows the Applicability to be entered (i.e., ≥25% RTP) prior to performing the Surveillance consistent with CTS 4.0.D allowance of CTS 4.11.A.4. Therefore, the specific Specification 4.0.D allowance of CTS 4.11.A.4 is not necessary and has been deleted.

L.2 CTS 4.11.A.3, which requires the APLHGRs to be verified to be within the limits initially and every 12 hours when operating at a LIMITING CONTROL ROD PATTERN, is proposed to be deleted. A LIMITING CONTROL ROD PATTERN is currently defined as operating on a power distribution limit such as APLHGR. This condition is extremely unlikely and the Surveillance would seldom be required. Additionally, the initial Surveillance is superfluous as it would not be evident that a LIMITING CONTROL ROD PATTERN has been achieved until the Surveillance is performed. Therefore, the Surveillance Frequency has been deleted.

RELOCATED SPECIFICATIONS

None

3.11 - LIMITING CONDITIONS FOR OPERATION

C. MINIMUM CRITICAL POWER RATIO

LCO 3.2.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR operating limit specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: A.2

~~OPERATIONAL MODE 1,~~ when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

ACTION A With MCPR less than the applicable MCPR operating limit as determined for one of the conditions specified in the CORE OPERATING LIMITS REPORT: LA.1

1. ~~Initiate corrective ACTION within 15 minutes, and~~
2. Restore MCPR to within the required limit within 2 hours.

ACTION B With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

C. MINIMUM CRITICAL POWER RATIO

SR 3.2.2.1 MCPR shall be determined to be equal to or greater than the applicable MCPR operating limit specified in the CORE OPERATING LIMITS REPORT.

1. At least once per 24 hours,
2. Within 12 hours after ~~completion of a~~ THERMAL POWER ~~increase of at least~~ ~~15%~~ ≥ 25% of RATED THERMAL POWER, and L.1
3. ~~Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.~~ L.2
4. ~~The provisions of Specification 4.0.D are not applicable.~~ L.1

Add proposed SR 3.2.2.2

M.1

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretation). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 The Applicability for CTS 3/4.11.C is "OPERATIONAL MODE 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER." With THERMAL POWER ≥ 25% RTP, the unit will always be in MODE 1. Therefore, it is unnecessary to state "OPERATIONAL MODE 1" in the Applicability of CTS 3/4.11.C (ITS 3.2.2).

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS 4.11.C requires the verification of the MCPR operating limits to be performed as specified in the COLR. Proposed ITS SR 3.2.2.2 specifies that the MCPR limits must be determined within 72 hours after each completion of ITS SR 3.1.4.1, SR 3.1.4.2, and SR 3.1.4.4 (control rod scram testing). This is a new requirement that assigns specific Surveillance Frequencies to the current practice for determining the MCPR Operating Limits based on Technical Specification Scram Speeds as specified in the COLR. This change is consistent with BWR ISTS, NUREG-1433, Revision 1, and its incorporation is necessary to ensure that MCPR limits are appropriately updated after scram time testing is complete. Since this change imposes added restraints on plant operation, it is considered a more restrictive change.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The requirement in CTS 3.11.C, ACTION 1, to "initiate corrective action within 15 minutes," to restore the limit is proposed to be relocated to the Bases in the form of a discussion that "prompt action" should be taken to restore the parameter to within limits. Immediate action may not always be the conservative method to assure safety. The ITS 3.2.2 ACTION A two hour completion time for restoration allows appropriate actions to be evaluated by the operator and

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.1 (cont'd) completed in a timely manner. As such, the relocated requirement is not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

"Specific"

L.1 CTS 4.11.C.2 is proposed to be changed to eliminate confusion as to how often the current Surveillance is required (e.g., after every 15% power change or at the end of any single power increase greater than 15%). Verifying the parameter within 12 hours of reaching or exceeding 25% RTP will generally require that the Surveillance be performed sooner than the CTS 4.11.C.2 requirement of "after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER," but would also reduce the number of times the Surveillance must be conducted during a startup if it is conducted after every 15% power change. A single verification is considered sufficient during initial startup considering the large inherent margin to operating limits at low power levels. Following the initial verification, the Surveillance is performed every 24 hours to identify any trends in these parameters that may lead to long term noncompliance. In addition, this change allows the applicability to be entered prior to performing the surveillance, consistent with the CTS 4.0.D allowance of CTS 4.11.C.4. Therefore, the specific Specification 4.0.D allowance of CTS 4.11.C.4 is not necessary and has been deleted.

L.2 CTS 4.11.C.3, which requires the MCPR to be verified to be within the limit initially and every 12 hours when operating at a LIMITING CONTROL ROD PATTERN, is proposed to be deleted. A LIMITING CONTROL ROD PATTERN is currently defined as operating on a power distribution limit, such as MCPR, the condition is extremely unlikely and the Surveillance would seldom be required. Additionally, the initial Surveillance is superfluous as it would not be evident that a LIMITING CONTROL ROD PATTERN has been achieved until the Surveillance is performed. Therefore, the Surveillance Frequency has been deleted.

RELOCATED SPECIFICATIONS

None

ITS 3.2.3

3.11 - LIMITING CONDITIONS FOR OPERATION

LCO 3.2.3 D. LINEAR HEAT GENERATION RATE

The LINEAR HEAT GENERATION RATE (LHGR) for each type of fuel shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY:

A.2 OPERATIONAL MODE 1 when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

ACTION A With a LHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT:

LA.1 1. ~~Initiate corrective ACTION within 15 minutes, and~~

2. Restore the LHGR to within the required limit within 2 hours.

ACTION B With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

D. LINEAR HEAT GENERATION RATE

SR 3.2.3.1 The LHGR shall be determined to be equal to or less than the limit:

1. At least once per 24 hours,

2. Within 12 hours after ~~completion of a~~ THERMAL POWER ~~increase of at least 15%~~ ≥25% of RATED THERMAL POWER, and L.1

3. ~~Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for LHGR.~~ L.2

4. ~~The provisions of Specification 4.0.D are not applicable.~~ L.1

Page 1 of 1

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretation). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 The Applicability for CTS 3/4.11.D is "OPERATIONAL MODE 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER." With THERMAL POWER ≥ 25% RTP, the unit will always be in MODE 1. Therefore, it is unnecessary to state "OPERATIONAL MODE 1" in the Applicability of CTS 3/4.11.D (ITS 3.2.3).

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The requirement in CTS 3.11.D ACTION 1 to "initiate corrective action within 15 minutes" to restore the limit is proposed to be relocated to the Bases in the form of a discussion that "prompt action" should be taken to restore the parameter to within the limits. Immediate action may not always be the conservative method to assure safety. The ITS 3.2.3 ACTION A 2 hour Completion Time for restoration allows appropriate actions to be evaluated by the operator and completed in a timely manner. As such, the relocated requirement is not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

"Specific"

L.1 CTS 4.11.D.2 is proposed to be changed to eliminate confusion as to how often the current Surveillance is required (e.g., after every 15% power change or at the end of any single power increase greater than 15%). Verifying the parameter within 12 hours of reaching or exceeding 25% RTP will generally require that the Surveillance be performed sooner than the CTS 4.11.D.2 requirement of "after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER," but would also reduce the number of times the Surveillance must be conducted during a startup if it is conducted after every 15% power change. A single verification is considered sufficient during initial startup considering the large inherent margin to operating limits at low power levels. Following the initial verification, the Surveillance is performed every 24 hours to identify any trends in these parameters that may lead to long term noncompliance. In addition, this change allows the Applicability to be entered (i.e., ≥25% RTP) prior to performing the Surveillance consistent with CTS 4.0.D allowance of CTS 4.11.D.4. Therefore, the specific Specification 4.0.D allowance of CTS 4.11.D.4 is not necessary and has been deleted.

L.2 CTS 4.11.D.3, which requires the LHGRs to be verified to be within the limits initially and every 12 hours when operating at a LIMITING CONTROL ROD PATTERN, is proposed to be deleted. A LIMITING CONTROL ROD PATTERN is currently defined as operating on a power distribution limit such as LHGR, the condition is extremely unlikely and the Surveillance would seldom be required. Additionally, the initial Surveillance is superfluous as it would not be evident that a LIMITING CONTROL ROD PATTERN has been achieved until the Surveillance is performed. Therefore, the Surveillance Frequency has been deleted.

RELOCATED SPECIFICATIONS

None

3.11 - LIMITING CONDITIONS FOR OPERATION

B. TRANSIENT LINEAR HEAT GENERATION RATE

APRM GAIN and Setpoint

A.2

The TRANSIENT LINEAR HEAT GENERATION RATE (TLHGR) shall be maintained such that the FUEL DESIGN

LCO 3.2.4.a LIMITING RATIO for CENTERLINE MELT (FDLRC)[a] is less than or equal to 1.0.

LCO 3.2.4.b Where FDLRC is equal to:

LCO 3.2.4.c

$$\frac{(LHGR)(1.2)}{(TLHGR)(FRTP)}$$

A.3

APPLICABILITY:

A.4

OPERATIONAL MODE 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

ACTION A

With FDLRC greater than 1.0, initiate corrective ACTION within 15 minutes and within 6 hours either:

LA.1

1. Restore FDLRC to less then or equal to 1.0, or

   Allowable Value

2. Adjust the flow biased APRM setpoints specified in Specifications 2.2.A and 3.2.E by 1/FDLRC, or

   A.5

   L.1

3. Adjust[b] each APRM gain such that the APRM readings are ≥100 times the FRACTION OF RATED THERMAL POWER (FRTP) times FDLRC.

ACTION B

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

B. TRANSIENT LINEAR HEAT GENERATION RATE

The value of FDLRC[a] shall be verified:

SR 3.2.4.1

1. At least once per 24 hours,

   L.2

   ≥ 25 %

2. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

SR 3.2.4.2

3. Initially and at least once per 12 hours when the reactor is operating with FDLRC greater than or equal to 1.0.

4. The provisions of Specification 4.0.D are not applicable.

   L.2

each required APRM Flow Biased Neutron Flux-High Function Allowable Value shall be modified by the lesser of 1/FDLRC or FRTP/MFLPD; or each required APRM gain shall be adjusted such that the APRM readings are ≥ 100 times the higher of FRTP times FDLRC or of MFLPD.

LA.2

A.2

A.5

LCO 3.2.4.a

a For GE fuel, MFLPD/FRTP is substituted for FDLRC. Adjustments are based on the lowest APRM setpoint or highest APRM reading resulting from the two limits.

L.1

b Provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretation). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 The CTS 3.11.B Action (with footnote a) provides an allowance to adjust the flow biased APRM setpoints (changed to Allowable Value as described in Discussion of Change A.5 below) or to adjust each APRM gain when FDLRC or MFLPD/FRTP is greater than 1.0. ITS 3.2.4 maintains this allowance in ACTION A, but also provides this allowance in the actual LCO. This is acceptable since CTS 3.11.B allows continued operation for an unlimited amount of time (i.e., FDLRC and MFLPD/FRTP do not have to be restored to ≤ 1.0) with the APRM setpoints or gains adjusted. Therefore, this presentation preference is considered administrative. In addition, the title of the Specification has been changed from Transient LHGR in CTS 3/4.11.B to APRM Gain and Setpoint in ITS 3.2.4, to reflect these new LCO additions.

A.3 The FUEL DESIGN LIMITING RATIO FOR CENTERLINE MELT (FDLRC) is defined in the CTS 1.0, Definitions Section, and is being retained in definitions in ITS Section 1.1. CTS 3.11.B defines FDLRC consistent with the current and proposed definitions, except that in CTS 3.11.B TLHGR is not indicated as a limit. However, based on the current methodology and definition, TLHGR is actually a limit. Since one purpose of the Definition Section is to minimize repetition so that requirements can be clearly defined it has been decided not to repeat the definition of FDLRC in ITS 3.2.4. The removal of this repeated definition is considered administrative since no requirements are being changed. This change is consistent with the BWR ISTS, NUREG-1433, Rev. 1.

A.4 The Applicability for CTS 3/4.11.B is "OPERATIONAL MODE 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER." With THERMAL POWER ≥ 25% RTP, the unit will always be in MODE 1. Therefore, it is unnecessary to state "OPERATIONAL MODE 1" in the Applicability of CTS 3/4.11.B (ITS 3.2.4).

A.5 The reference to the Trip Setpoint of the APRM Flow Biased Neutron Flux — High trip in CTS 3.11.B ACTION 2, has been changed to Allowable Value since the Trip Setpoint is not included in the ITS (see Discussion of Change in the RPS Specification, ITS 3.3.1.1). As such, this change is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The requirement in the CTS 3.11.B ACTION to "initiate corrective action within 15 minutes" to restore the limit is proposed to be relocated to the Bases in the form of a discussion that "prompt action" should be taken to restore the parameter to within the limits. Immediate action may not always be the conservative method to assure safety. The ITS 3.2.4 ACTION A 6 hour Completion Time for restoration allows appropriate actions to be evaluated by the operator and completed in a timely manner. As such, the relocated requirement is not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

LA.2 The detail in CTS 3.11.B footnote a that for GE fuel, MFLPD is substituted for FDLRC is proposed to be relocated to the Bases. ITS LCO 3.2.4.a requires FDLRC and the ratio of MFLPD to fraction of RTP (FRTP) to be less than or equal to 1.0. This will require that the most limiting value of FDLRC for Siemens fuel and the most limiting value of MFLPD/FRTP for GE fuel be less than or equal to 1.0. The Bases provides the details how each vendor protects the core from local peaking. As a result, this detail is not necessary to be included in the Technical Specifications to ensure the core is protected from local peaking. As such, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

"Specific"

L.1 CTS 3.11.B ACTION 3, including footnote b, allows the APRM gain to be adjusted so that the APRM readings are greater than or equal to 100% times FRTP times FDLRC, provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel. The posting of the adjustment in the control room is

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 (cont'd) not necessary to be described in Technical Specifications. This requirement is essentially an "operator aid" to remind the operators that an adjustment has been made. This requirement is not necessary in the Technical Specifications to ensure power is maintained within the limit allowed by the Operating License. Operators are required by 10 CFR 55 to comply with the Operating License. Therefore, this requirement has been deleted from Technical Specifications.

L.2 CTS 4.11.B.2 (the verification of FDLRC or MFLPD to FRTP) is proposed to be changed to eliminate confusion as to how often the current Surveillance is required (e.g., after every 15% power change or at the end of any single power increase greater than 15%). Verifying the parameter within 12 hours of reaching or exceeding 25% RTP will generally require that the Surveillance be performed sooner than the CTS 4.11.B.2 requirement of "after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER," but would also reduce the number of times the Surveillance must be conducted during a startup if it is currently conducted after every 15% power change. A single verification is considered sufficient during initial startup considering the large inherent margin to APLHGR, LHGR, and MCPR operating limits at low power levels. At higher power levels, core peaking is reduced and therefore the need to adjust the APRMs or Flow Biased Scram Setpoints is reduced. Following the initial verification, the Surveillance is performed every 24 hours to identify any trends in these parameters that may lead to long term noncompliance. However, since core nuclear instrumentation is monitored, any anomaly will be detected and corrected between required Surveillances during a power ascension. In addition, this change allows the Applicability to be entered prior to performing the Surveillance, consistent with the current Specification 4.0.D allowance of CTS 4.11.B.4. Therefore, the specific Specification 4.0.D allowance of CTS 4.11.B.4 is not necessary and has been deleted.

RELOCATED SPECIFICATIONS

None

The Bases of the current Technical Specifications for this section (pages B 3/4.11-1 through B 3/4.11-4) have been completely replaced by revised Bases that reflect the format and applicable content of the Quad Cities 1 and 2 ITS Section 3.2, consistent with the BWR ISTS, NUREG-1433, Rev. 1. The revised Bases are as shown in the Quad Cities 1 and 2 ITS Bases.

<CTS>

3.2 POWER DISTRIBUTION LIMITS

3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

<3.11.A> LCO 3.2.1 All APLHGRs shall be less than or equal to the limits specified in the COLR.

<Appl 3.11.A> APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

<3.11.A Act>

<3.11.A Act>

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any APLHGR not within limits.	A.1 Restore APLHGR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

<4.11.A>

SURVEILLANCE	FREQUENCY
SR 3.2.1.1 Verify all APLHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 25% RTP **AND** 24 hours thereafter

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.2.1 - AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1. There are no deviations from NUREG-1433, Revision 1, for proposed Specification 3.2.1.

<CTS>

3.2 POWER DISTRIBUTION LIMITS

3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

<3.11.C>

LCO 3.2.2 All MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR.

<Appl 3.11.C>

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.11.C Act> A. Any MCPR not within limits.	A.1 Restore MCPR(s) to within limits.	2 hours
<3.11.C Act> B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<4.11.C> SR 3.2.2.1 Verify all MCPRs are greater than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 25% RTP **AND** 24 hours thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.2.2.2 Determine the MCPR limits.	Once within 72 hours after each completion of SR 3.1.4.1 **AND** Once within 72 hours after each completion of SR 3.1.4.2 AND Once within 72 hours after each completion of SR 3.1.4.4

⟨DOC M.1⟩

1

TSTF-229

1. NUREG-1433, Revision 1, SR 3.2.2.2 requires the MCPR limit to be determined once within 72 hours after each completion of SR 3.1.4.1 and SR 3.1.4.2. SR 3.2.2.2 also needs to be performed after each completion of SR 3.1.4.4, which is the individual control rod scram test at high pressure, required after work that could affect the control rod scram speed. This change is necessary to ensure that the MCPR limits are appropriately updated after scram time testing is complete. This change is also consistent with TSTF-229, Rev. 0.

1

3.2 POWER DISTRIBUTION LIMITS

3.2.3 LINEAR HEAT GENERATION RATE (LHGR) (Optional)

⟨3.11.D⟩ LCO 3.2.3 All LHGRs shall be less than or equal to the limits specified in the COLR.

⟨Appl 3.11.D⟩ APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
⟨3.11.D Act⟩ A. Any LHGR not within limits.	A.1 Restore LHGR(s) to within limits.	2 hours
⟨3.11.D Act⟩ B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
⟨4.11.D⟩ SR 3.2.3.1 Verify all LHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter

1. This reviewer's type of note has been deleted. This is not meant to be retained in the final version of the plant specific submittal.

<CTS>

[1] [2] [4]

3.2 POWER DISTRIBUTION LIMITS

3.2.4 Average Power Range Monitor (APRM) Gain and Setpoints (Optional)

(FRTP)

[3]

<3.11.B> LCO 3.2.4

FDLRC and the ratio of MFLPD to

a. MFLPD shall be less than or equal to [1] Fraction of RTP; or [3] 1.0

b. Each required APRM setpoint specified in the COLR shall be made applicable; or

c. Each required APRM gain shall be adjusted such that the APRM readings are ≥ 100% times MFLPD.

the higher of FRTP times FDLRC or of

Flow Biased Neutron Flux - High Function Allowable Value shall be modified by the lesser of 1/FDLRC or FRTP/MFLPD

<Appl 3.11.8> [3] [4] APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

[3] [4]

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.11.B Act> A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	6 hours
<3.11.B Act> B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1 <4.11.B>	------------------NOTE------------------ Not required to be met if SR 3.2.4.2 is satisfied for LCO 3.2.4 ~~Item~~ o b or c ~~requirements.~~ LCO 3.2.4. 5 -- Verify ~~MFLPD is~~ FDLRC and the ratio of MFLPD to FRTP are 3 within limits.	Once within 12 hours after ≥ 25% RTP **AND** 24 hours thereafter
SR 3.2.4.2 <4.11.B>	------------------NOTE------------------ Not required to be met if SR 3.2.4.1 is satisfied for LCO 3.2.4 ~~Item~~ o a requirements. 5 -- Verify ~~APRM setpoints or gains are adjusted for the calculated MFLPD.~~	12 hours

each required: 3 4

a. APRM Flow Biased Neutron Flux - High Function Allowable Value is modified by less than or equal to the lesser of 1/FDLRC or FRTP/MFLPD; or

b. APRM gain is adjusted such that the APRM reading is ≥ 100% times the higher of FRTP times FDLRC or of MFLPD.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.2.4 - APRM GAIN AND SETPOINT

1. Typographical/grammatical error corrected.

2. This reviewer's type of note has been deleted. This is not meant to be retained in the final version of the plant specific submittal.

3. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description (Quad Cities uses both GE and Siemens fuel), or licensing basis description.

4. The APRM setpoint modification is not cycle specific. Therefore, references to the COLR have been deleted and the proper modification to the "setpoint" has been provided. This modification is consistent with the CTS and Bases. In addition, the word "setpoint" has been replaced with the name of the actual APRM Function that is being modified, consistent with similar statements in other places in the ITS. Also, the acronym "FRTP" has been defined in ITS 3.2.4.a consistent with the plant specific use for APRM gain adjustment. ITS SR 3.2.4.2 has been modified to reflect the changes made to the LCO.

5. Editorial change to be consistent with similar statements in other places in the ITS.

All changes are [1] unless otherwise indicated

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

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

Additionally, for General Electric fuel types, APLHGR limits are specified to ensure

APPLICABLE SAFETY ANALYSES

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

move to next page as indicated

Fuel design evaluations are performed to demonstrate that the 1% limit on the fuel cladding plastic strain and other fuel design limits described in Reference 1 are not exceeded during AOOs for operation with LHGRs up to the operating limit LHGR. APLHGR limits are equivalent to the LHGR limit for each fuel rod divided by the local peaking factor of the fuel assembly. APLHGR limits are developed as a function of exposure ~~and the various operating core flow and power states~~ to ensure adherence to fuel design limits during the limiting AOOs (Refs. ~~5, 6, and 7~~). (4) ~~Flow dependent APLHGR limits are determined using the three dimensional BWR simulator code (Ref. 8) to analyze slow flow runout transients. The flow dependent multiplier, MAPFAC$_f$, is dependent on the maximum core flow runout capability. The maximum runout flow is dependent on the existing setting of the core flow limiter in the Recirculation Flow Control System.~~

move to next page as indicated

~~Based on analyses of limiting plant transients (other than core flow increases) over a range of power and flow conditions, power dependent multipliers, MAPFAC$_p$, are also generated. Due to the sensitivity of the transient response to initial core flow levels at power levels below those at~~

(continued)

All changes are $\boxed{1}$ unless otherwise indicated

BASES

APPLICABLE SAFETY ANALYSES (continued)

A conservative multiplier is applied to the LHGR and APLHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR. For GE analyses only,

~~which turbine stop valve closure and turbine control valve fast closure scram trips are bypassed, both high and low core flow $MAPFAC_p$ limits are provided for operation at power levels between 25% RTP and the previously mentioned bypass power level. The exposure dependent APLHGR limits are reduced by $MAPFAC_p$ and $MAPFAC_f$ at various operating conditions to ensure that all fuel design criteria are met for normal operation and AOOs. A complete discussion of the analysis code is provided in Reference 9.~~

LOCA analyses are ~~then~~ performed to ensure that the ~~above determined~~ APLHGR limits are adequate to meet the PCT peak cladding temperature 2 and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using calculational models that are consistent with the requirements of 10 CFR 50, Appendix K. A complete discussion of the analysis code is provided in Reference ~~10~~ 1 and 5. The PCT following a postulated LOCA is a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within an assembly. For Siemens Power Corporation Fuel, APLHGR limits are typically set high enough such that the LHGR limits are more limiting than the APLHGR limits. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by ~~its~~ the minimum anticipated local peaking factor. ~~A conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.~~

2

For single recirculation loop operation, ~~the MAPFAC multiplier is limited to a maximum of 0.75 (Ref. 5)~~ a conservative multiplier is applied to the exposure dependent APLHGR limits for two loop operation (Ref. 6). This ~~maximum limit~~ additional limitation is due to the conservative analysis assumption of an earlier departure from nucleate boiling with one recirculation loop available, resulting in a more severe cladding heatup during a LOCA.

move from ASA first paragraph

move from ASA second paragraph

The APLHGR satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii).

For GE fuel types, the APLHGR limits also incorporate the results of the fuel design limits.

LCO

The APLHGR limits specified in the COLR are the result of the fuel design, DBA, and transient analyses. For two recirculation loops operating, the limit is ~~determined by multiplying the smaller of the $MAPFAC_p$ and $MAPFAC_f$ factors times the exposure dependent APLHGR limits~~ dependent on exposure. With only one recirculation loop in operation, in conformance with the requirements of LCO 3.4.1, "Recirculation Loops Operating,"

(continued)

BASES

a conservative multiplier 1

LCO (continued)

the limit is determined by multiplying the exposure dependent APLHGR limit by ~~the smaller of either $MAPFAC_p$, $MAPFAC_f$, and 0.75, where 0.75 has been~~ determined by a specific single recirculation loop analysis (Ref. ~~8~~ 6). 1

APPLICABILITY

The APLHGR limits are primarily derived from fuel design evaluations and LOCA and transient analyses that are assumed to occur at high power levels. ~~Design calculations~~ Studies (Ref. 7) 1 and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels ≤ 25% RTP, the reactor is operating with substantial margin to the APLHGR limits; thus, this LCO is not required.

ACTIONS

A.1

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

B.1

If the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on 2

(continued)

BASES

ACTIONS

B.1 (continued)

operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

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

E 1

REFERENCES

1. NEDO-24011-P-A "General Electric Standard Application for Reactor Fuel" (~~latest approved version~~ as specified in Technical Specification 5.6.5). 1

1 u

2. FSAR, Chapter [4].

3. FSAR, Chapter [6]. 3

4. FSAR, Chapter [15].

~~5. [Plant specific single loop operation].~~

~~6. [Plant specific load line limit analysis].~~

~~7. [Plant Specific Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvements (ARTS) Program].~~

~~8. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.~~

1

5. EMF-94-217(NP), Revision 1, "Boiling Water Reactor Licensing Methodology Summary," November 1995

(continued)

BASES

REFERENCES (continued)

9. ~~NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.~~

10. ~~[Plant specific loss of coolant accident analysis].~~

6. UFSAR, Section 6.3.3.2.2.4

1

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.2.1 - AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

2. Editorial changes made to be consistent with similar statements in other places in the Bases.

3. The brackets have been removed and the proper plant specific value/information included.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND 1

MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR Safety Limit (SL) is set such that 99.9% of the fuel rods are expected to avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2). The operating limit MCPR is established to ensure that no fuel damage results during anticipated operational occurrences (AOOs). Although fuel damage does not necessarily occur if a fuel rod actually experienced boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion. 7 2

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

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the AOOs to establish the operating limit MCPR are presented in References 2, 3, 4, 5, 6, 7, 8, and ~~8~~ 9. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained. 1

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power 1

(continued)

All changes are [1] unless otherwise indicated

BASES

as identified in UFSAR, Chapter 15 (Ref. 5)

APPLICABLE SAFETY ANALYSES (continued)

state ($MCPR_f$ ~~and $MCPR_p$, respectively~~) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency ~~(Refs. 6, 7, and 8)~~. Flow dependent MCPR limits are determined by steady state thermal hydraulic methods with key physics response inputs benchmarked using the three dimensional BWR simulator code (Ref. 8) to analyze slow flow runout transients. The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.

and a multichannel thermal hydraulic code (Ref. 9)

Insert ASA

~~Power dependent MCPR limits ($MCPR_p$) are determined mainly by the one dimensional transient code (Ref. 10). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scrams are bypassed, high and low flow $MCPR_p$ operating limits are provided for operating between 25% RTP and the previously mentioned bypass power level.~~

The MCPR satisfies Criterion 2 of ~~the NRC Policy Statement~~. 10 CFR 50.36(c)(2)(ii)

LCO

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

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a ~~minimum~~ low recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Statistical analyses indicate that the nominal value of the initial MCPR expected at 25% RTP is > 3.5. Studies of the variation of limiting transient behavior have been performed over the range of power and flow conditions. These studies encompass the range of key actual plant parameter values important to typically limiting transients. The results of these studies demonstrate that a margin is expected between performance and the MCPR requirements, and that margins increase as

(continued)

$\boxed{1}$

Insert ASA

on a cycle-specific basis. For core flows less than rated, the established MCPR operating limit is adjusted to provide protection of the MCPR SL in the event of an uncontrolled recirculation flow increase to the physical limit of the pump. Protection is provided for manual and automatic flow control by applying appropriate flow dependent MCPR operating limits. The MCPR operating limit for a given flow state is the greater of the rated conditions MCPR operating limit or the flow dependent MCPR operating limit. For automatic flow control, in addition to protecting the MCPR SL during the flow run-up event, protection is provided by the flow dependent MCPR operating limit to prevent exceeding the rated flow MCPR operating limit during an automatic flow increase to rated core flow.

BASES

APPLICABILITY (continued)

power is reduced to 25% RTP. This trend is expected to continue to the 5% to 15% power range when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor provides rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < 25% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required.

ACTIONS

A.1

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

B.1

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

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. It is compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and

(continued)

All changes are ☐1 unless otherwise indicated

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1 (continued)

recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

1

The MCPR limit, including the scram insertion times for rated and off-rated flow conditions, are contained in the COLR.

SR 3.2.2.2

Because the transient analysis takes credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. SR 3.2.2.2 determines ~~the value of τ, which is a measure of~~ the actual scram speed distribution and compares it with the assumed distribution. The MCPR operating limit is then determined based on ~~an interpolation between~~ either the applicable limits ~~for Option A~~ associated with the scram times of LCO 3.1.4, "Control Rod Scram Times," ~~and Option B~~ or the realistic scram times analyses. The ~~parameter τ~~ determination must be ~~determined~~ performed once within 72 hours after each set of scram time tests required by SR 3.1.4.1 and SR 3.1.4.2 and SR 3.1.4.4 because the effective scram speed distribution may change during the cycle or after maintenance that could affect scram times. The 72 hour Completion Time is acceptable due to the relatively minor changes in τ the actual scram speed distribution expected during the fuel cycle.

3 TSTF-229

TSTF-229

REFERENCES

1. NUREG-0562, June 1979.
2. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (~~latest approved version~~ as specified in Technical Specification 5.6.5).
3. UFSAR, Chapter ~~[4]~~.
4. UFSAR, Chapter ~~[6]~~.
5. UFSAR, Chapter ~~[15]~~.
6. ~~[Plant specific single loop operation].~~
7. ~~[Plant specific load line limit analysis].~~

2 4

(continued)

BASES

REFERENCES (continued)

8. ~~[Plant specific Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvements (ARTS) Program].~~

9. ~~NEDO-30130-A, "Steady State Nuclear Methods," May 1985.~~

10. ~~NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.~~

1

6. EMF-94-217(NP), Revision 1, "Boiling Water Reactor Licensing Methodology Summary," November 1995

7. NFSR-0091, Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods, Commonwealth Edison Topical Report (as specified in Technical Specification 5.6.5)

8. XN-NF-80-19 (P)(A), Volume 1, Exxon Nuclear Methodology for Boiling Water Reactors, Neutronic Methods for Design and Analysis (as specified in Technical Specification 5.6.5)

9 XN-NF-80-19 (P)A, Volume 3, Exxon Nuclear Methodology for Boiling Water Reactors THERMEX Thermal Limits Methodology Summary Description, (as specified in Technical Specification 5.6.5)

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.2.2 - MINIMUM CRITICAL POWER RATIO

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

2. Typographical error corrected.

3. Changes have been made to reflect those changes made to the Specification.

4. The brackets have been removed and the proper plant specific value/information included.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR) ~~(Optional)~~ 1

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences (AOOs). Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure, or inability to cool the fuel does not occur during the normal operations and anticipated operating conditions identified in References 1 and 2. 2

APPLICABLE SAFETY ANALYSES

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

~~a.~~ Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet~~; and~~.

~~b. Severe overheating of the fuel rod cladding caused by inadequate cooling.~~ 3

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

Fuel design evaluations have been performed and demonstrate that the ~~[1%]~~ fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to 3

(continued)

2 while still remaining within the AOO limits

BASES

APPLICABLE SAFETY ANALYSES (continued)

the operating limit specified in the COLR. The analysis also includes allowances for short term transient ~~operation~~ 2 excursions above the operating limit ~~to account for AOOs~~, plus an allowance for densification power spiking.

The LHGR satisfies Criterion 2 of ~~the NRC Policy Statement~~ 2 10 CFR 50.36 (c)(2)(ii).

LCO

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

APPLICABILITY

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

ACTIONS

A.1

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

B.1

If the LHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER is reduced to < 25% RTP within 4 hours. The allowed

(continued)

BASES

ACTIONS

B.1 (continued)

Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER TO < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1 5

are They are 4

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

LHGR

2

REFERENCES

a

~~1.~~ FSAR, ~~Section [ ].~~ Chapter 4 3

2. FSAR, ~~Section [ ].~~ Chapter 15

3. NUREG-0800, Section II.A.2(g), Revision 2, July 1981.

4.2 2

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.2.3 - LINEAR HEAT GENERATION RATE

1. Changes have been made to reflect those changes made to the Specification.

2. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

3. The brackets have been removed and the proper plant specific information/value has been provided.

4. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

1

2

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 Average Power Range Monitor (APRM) Gain and Setpoints (Optional) 1

BASES

BACKGROUND

The OPERABILITY of the APRMs and their setpoints is an initial condition of all safety analyses that assume rod insertion upon reactor scram. Applicable GDCs are GDC 10, "Reactor Design," GDC 13, "Instrumentation and Control," GDC 20, "Protection System Functions," and GDC 23, "Protection against Anticipated Operation Occurrences" (Ref. 1). This LCO is provided to require the APRM gain or APRM flow biased scram setpoints to be adjusted when operating under conditions of excessive power peaking to maintain acceptable margin to the fuel cladding integrity Safety Limit (SL) and the fuel cladding 1% plastic strain limit.

general design criteria are discussed in UFSAR, Sections 3.1.2.1, 3.1.3.2, 3.1.3.4, 3.1.3.5 and 3.1.4.8

3

1

Neutron Flux - High Function Allowable Value (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b

The condition of excessive power peaking is determined by the ratio of the actual power peaking to the limiting power peaking at RTP. This ratio is equal to the ratio of the core limiting MFLPD to the Fraction of RTP (FRTP), where FRTP is the measured THERMAL POWER divided by the RTP. Excessive power peaking exists when:

For General Electric (GE) fuel,

3

Insert BKGD

$$\frac{MFLPD}{FRTP} > 1,$$

indicating that MFLPD is not decreasing proportionately to the overall power reduction, or conversely, that power peaking is increasing. To maintain margins similar to those at RTP conditions, the excessive power peaking is compensated by a gain adjustment on the APRMs or adjustment of the APRM setpoints. Either of these adjustments has effectively the same result as maintaining MFLPD less than or equal to FRTP and thus maintains RTP margins for APLHGR and MCPR.

1

Neutron Flux - High Function Allowable Value

1

FDLRC and the ratio of

1

modification

1

to FRTP

1

1.0

3

and LHGR

3

Flow Biased Neutron Flux - High Function Allowable Value

1

The normally selected APRM setpoints position the scram above the upper bound of the normal power/flow operating region that has been considered in the design of the fuel rods. The setpoints are flow biased with a slope that approximates the upper flow control line, such that an approximately constant margin is maintained between the flow biased trip level and the upper operating boundary for core flows in excess of about 45% of rated core flow. In the range of infrequent operations below 45% of rated core flow,

Allowable Value is

1

Adjustments are based on the lowest APRM Neutron Flux - High Function Allowable Value or highest APRM reading resulting from the two methods (GE or Siemens)

3

(continued)

$\boxed{3}$

Insert BKGD

For Siemens (SPC) fuel, the condition of excessive power peaking is determined by Fuel Design Limit Ratio for Centerline Melt (FDLRC), which is defined as:

$$FDLRC = \frac{(LHGR)(1.2)}{(TLHGR)(FRTP)} \;;$$

where LHGR is the Linear Heat Generation Rate, FRTP is the Fraction of Rated Thermal Power, and TLHGR is the Transient Linear Heat Generation Rate limit. The TLHGR limit is specified in the COLR and protects against fuel centerline melting and the fuel cladding 1% plastic strain during transient conditions throughout the life of the fuel.

BASES

[3]

BACKGROUND (continued)

[1] Allowable Value is

the margin to scram is reduced because of the nonlinear core flow versus drive flow relationship. The normally selected APRM ~~setpoints~~ are supported by the analyses presented in References ~~1 and~~ 2 that concentrate on events initiated from rated conditions. Design experience has shown that minimum deviations occur within expected margins to operating limits (APLHGR ~~and~~ MCPR), at rated conditions for normal power distributions. However, at other than rated conditions, control rod patterns can be established that significantly reduce the margin to thermal limits. Therefore, the ~~flow biased~~ APRM ~~scram setpoints~~ may be reduced during operation when the combination of THERMAL POWER and MFLPD indicates an excessive power peaking distribution.

[3] , and LHGR

[3] (3)

Flow Biased Neutron Flux - High Function Allowable Value [1]

FDLRC or [1]

[3]

~~The APRM neutron flux signal is also adjusted to more closely follow the fuel cladding heat flux during power transients. The APRM neutron flux signal is a measure of the core thermal power during steady state operation. During power transients, the APRM signal leads the actual core thermal power response because of the fuel thermal time constant. Therefore, on power increase transients, the APRM signal provides a conservatively high measure of core thermal power. By passing the APRM signal through an electronic filter with a time constant less than, but approximately equal to, that of the fuel thermal time constant, an APRM transient response that more closely follows actual fuel cladding heat flux is obtained, while a conservative margin is maintained. The delayed response of the filtered APRM signal allows the flow biased APRM scram levels to be positioned closer to the upper bound of the normal power and flow range, without unnecessarily causing reactor scrams during short duration neutron flux spikes. These spikes can be caused by insignificant transients such as performance of main steam line valve surveillances or momentary flow increases of only several percent.~~

APPLICABLE SAFETY ANALYSES

[3] (3) , and LHGR

The acceptance criteria for the APRM gain or setpoint adjustments are that acceptable margins (to APLHGR ~~and~~ MCPR) be maintained to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

[3] (4) (1) (2) [3]

FSAR safety analyses (Refs. ~~2 and 3~~) concentrate on the rated power condition for which the minimum expected margin to the operating limits (APLHGR ~~and~~ MCPR) occurs.

(3) [3]

(continued)

All changes are [1] unless otherwise indicated

[2] [3]

BASES

APPLICABLE SAFETY ANALYSES (continued)

[3]

LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limit the initial margins to these operating limits at rated conditions so that specified acceptable fuel design limits are met during transients initiated from rated conditions. At initial power levels less than rated levels, the margin degradation of either the APLHGR or the MCPR during a transient can be greater than at the rated condition event. This greater margin degradation during the transient is primarily offset by the larger initial margin to limits at the lower than rated power levels. However, power distributions can be hypothesized that would result in reduced margins to the pre-transient operating limit. When combined with the increased severity of certain transients at other than rated conditions, the SLs could be approached. At substantially reduced power levels, highly peaked power distributions could be obtained that could reduce thermal margins to the minimum levels required for transient events. To prevent or mitigate such situations, either the APRM gain is adjusted upward by the ratio of the core limiting MFLPD to the FRTP, or the flow biased APRM scram level is required to be reduced by the ratio of FRTP to the core limiting MFLPD. Either of these adjustments effectively counters the increased severity of some events at other than rated conditions by proportionally increasing the APRM gain or proportionally lowering the flow biased APRM scram setpoints, dependent on the increased peaking that may be encountered.

and LCO 3.2.3, "Linear Heat Generation Rate (LHGR),"

, or the LHGR

[3]

fuel design limits and MCPR SL

the higher of the

Flow Biased Neutron Flux – High Function Allowable Value

value of FDLRC or the ratio of the core limiting

by the lesser of either the reciprocal of the core limiting FDLRC or

The APRM gain and setpoints satisfy Criteria 2 and 3 of the NRC Policy Statement.

[3]

10 CFR 50.36(c)(2)(ii)

LCO

Meeting any one of the following conditions ensures acceptable operating margins for events described above:

a. Limiting excess power peaking;

b. Reducing the APRM flow biased neutron flux upscale scram setpoints by multiplying the APRM setpoints by the ratio of FRTP and the core limiting value of MFLPD; or

the lesser of either 1/FDLRC or

(continued)

All changes are [1] unless otherwise indicated

BASES

LCO (continued)

c. Increasing APRM gains to cause the APRM to read greater than 100 times MFLPD (in %). This condition is to account for the reduction in margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

[3] or equal to; (%) [4]

For GE fuel, [3] the higher of the core limiting value of FDLRC times FRTP or the core limiting

MFLPD is the ratio of the limiting LHGR to the LHGR limit for the specific bundle type. As power is reduced, if the design power distribution is maintained, MFLPD is reduced in proportion to the reduction in power. However, if power peaking increases above the design value, the MFLPD is not reduced in proportion to the reduction in power. Under these conditions, the APRM gain is adjusted upward or the APRM flow biased scram setpoints are reduced accordingly. When the reactor is operating with peaking less than the design value, it is not necessary to modify the APRM flow biased scram setpoints. Adjusting APRM gain or setpoints is equivalent to MFLPD less than or equal to FRTP as stated in the LCO.

is; and FDLRC; are; and FDLRC; are

[3] For Siemens fuel FDLRC times FRTP is the ratio of the LHGR times 1.2 to TLHGR

Neutron Flux-High Function Allowable Value

modifying the APRM Flow Biased Neutron Flux-High Function Allowable Value

maintaining FDLRC and the ratio of; less than or equal to 1.0

For compliance with LCO Item b (APRM setpoint adjustment) or Item c (APRM gain adjustment), only APRMs required to be OPERABLE per LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," are required to be adjusted. In addition, each APRM may be allowed to have its gain or setpoints adjusted independently of other APRMs that are having their gain or setpoints adjusted.

3.2.4; modification; LCO 3.2.4.; modified or; Function 2.b; or Allowable Value modified; Flow Biased Neutron Flux-High Function Allowable Value

APPLICABILITY

The MFLPD limit, APRM gain adjustment, and APRM flow biased scram and associated setdowns are provided to ensure that the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit are not violated during design basis transients. As discussed in the Bases for LCO 3.2.1 and LCO 3.2.2, sufficient margin to these limits exists below 25% RTP and, therefore, these requirements are only necessary when the reactor is operating at ≥ 25% RTP.

to FRTP; or; Neutron Flux-High Function Allowable Value; The FDLRC or the ratio of; modification; [3] LCO 3.2.3

ACTIONS

A.1

If the APRM gain or setpoints are not within limits while the MFLPD has exceeded FRTP, the margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit

is; FDLRC or the ratio of MFLPD to FRTP exceed 1.0; Flow Biased Neutron Flux-High Function Allowable Value

(continued)

All changes are [1] unless otherwise indicated

BASES

ACTIONS

A.1 (continued)

may be reduced. Therefore, prompt action should be taken to restore ~~the MFLPD~~ FDLRC and the ratio of MFLPD to FRTP to within its required limit or make acceptable APRM adjustments such that the plant is operating within the assumed margin of the safety analyses.

The 6 hour Completion Time is normally sufficient to restore either ~~the MFLPD~~ FDLRC and the ratio of MFLPD to FRTP to within limits or the APRM gain or to adjust ~~setpoints~~ APRM Flow Biased Neutron Flux-High Function Allowable Value to within limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LCO not met.

modify the

B.1

If ~~MFLPD~~ FDLRC and the ratio of MFLPD to FRTP cannot be restored or the APRM gain or Flow Biased Neutron Flux-High Allowable Value to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER is reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2

~~The MFLPD~~ FDLRC and the ratio of MFLPD to FRTP is ~~required to be calculated~~ and compared to 1.0 ~~FRTP~~ or APRM gain adjusted or ~~setpoints~~ modified to ensure that the reactor is operating within the assumptions of the safety analysis. These SRs are only required to determine ~~the MFLPD~~ and, assuming ~~MFLPD is greater than FRTP~~ either exceeds 1.0, the appropriate gain or ~~setpoint~~ APRM, and are not intended to be a CHANNEL FUNCTIONAL TEST for the APRM gain or ~~flow biased neutron flux scram circuitry~~ APRM Flow Biased Neutron Flux—High Function. Insert SR [4] The 24 hour Frequency of SR 3.2.4.1 is chosen to coincide with the determination of other thermal limits, specifically those for the APLHGR (LCO 3.2.1), MCPR (LCO 3.2.2) and LHGR (LCO 3.2.3) [3]. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to APLHGR, MCPR and LHGR operating limits at low power levels.

[4] determine

(continued)

4

Insert SR

SR 3.2.4.1 and SR 3.2.4.2 have been modified by Notes, which clarify that the respective SR does not have to be met if the alternate requirement demonstrated by the other SR is satisfied.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2 (continued)

is d because 4

The 12 hour Frequency of SR 3.2.4.2 ~~requires a more frequent verification than if MFLPD is less than or equal to fraction of rated power (FRP).~~ When MFLPD is greater than FRP, more rapid changes in power distribution are typically expected.

1.0

either FDLRC or the ratio of MFLPD to FRTP 1

REFERENCES 3

1. 10 CFR 50, Appendix A, GDC 10, GDC 13, GDC 20, and GDC 23.

UFSAR, Sections 3.1.2.1, 3.1.3.2, 3.1.3.4, 3.1.3.5, and 3.1.4.8. 3

2. (U)FSAR, ~~Section [ ]~~. Chapter 15 5

~~3. FSAR, Section [/]~~ 3

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.2.4 - APRM GAIN AND SETPOINT

1. Changes have been made to reflect those changes made to the Specification.

2. Typographical/grammatical error corrected.

3. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

4. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

5. The brackets have been removed and the proper plant specific information/value has been provided.

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: SECTION 3.2 - POWER DISTRIBUTION LIMITS

ADMINISTRATIVE CHANGES
("A.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change involves reformatting, renumbering, and rewording the existing Technical Specifications. The reformatting, renumbering, and rewording process involves no technical changes to the existing Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analyses assumptions. This change is administrative in nature. Therefore, the change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - MORE RESTRICTIVE
("M.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change provides more stringent requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change 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 the methods governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The imposition of more restrictive requirements either has no impact on or increases the margin of plant safety. As provided in the discussion of the change, each change in this category is by definition, providing additional restrictions to enhance plant safety. The change maintains requirements within the safety analyses and licensing basis. Therefore, this change does not involve a significant reduction in a margin of safety.

"GENERIC" LESS RESTRICTIVE CHANGES:
RELOCATING DETAILS TO TECHNICAL SPECIFICATION BASES, UFSAR, TRM, OR OTHER PLANT CONTROLLED DOCUMENTS
("LA.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change relocates certain details from the Technical Specifications to the Bases, UFSAR, TRM, or other plant controlled documents. The Bases, UFSAR, TRM, and other plant controlled documents containing the relocated information will be maintained in accordance with 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specification Bases are subject to the change control provisions in the Administrative Controls Chapter of the ITS. The UFSAR is subject to the change control provisions of 10 CFR 50.71(e), and the plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Since any changes to the Bases, UFSAR, TRM, or other plant controlled documents will be evaluated per the requirements of the Bases Control Program in Chapter 5.0 of the ITS or 10 CFR 50.59, no increase (significant or insignificant) in the probability or consequences of an accident previously evaluated will be allowed. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change 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 a change in the methods governing normal plant operation. The proposed change will not impose or eliminate any requirements, and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the details to be transposed from the Technical Specifications to the Bases, UFSAR, TRM, or other plant controlled

"GENERIC" LESS RESTRICTIVE CHANGES:
RELOCATING DETAILS TO TECHNICAL SPECIFICATION BASES, UFSAR, TRM, OR OTHER PLANT CONTROLLED DOCUMENTS
("LA.x" Labeled Comments/Discussions)

3. (continued)

documents are the same as the existing Technical Specifications. Since any future changes to these details in the Bases, UFSAR, TRM, or other plant controlled documents will be evaluated per the requirements of 10 CFR 50.59, no reduction (significant or insignificant) in a margin of safety will be allowed. Based on 10 CFR 50.92, the existing requirement for NRC review and approval of revisions, to these details proposed for relocation, does not have a specific margin of safety upon which to evaluate. However, since the proposed change is consistent with the BWR ISTS, NUREG-1433, Rev. 1, approved by the NRC Staff, revising the Technical Specifications to reflect the approved level of detail ensures no significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.2.1 - AVERAGE PLANAR LINEAR HEAT GENERATION RATE

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The change to the Surveillance Frequency will require the verification of the APLHGR limit only once during low power operations with periodic reverification to identify trends. The APLHGR limit is used to verify the unit is operating within the initial assumptions of the safety analysis. Significant changes in this parameter are indicative of unanticipated operation, but are not, in themselves, identified as initiators of any previously analyzed accident. Therefore, the change in Frequency of the Surveillance will not significantly increase the probability of an accident previously identified. At low power, there are large inherent margins to the APLHGR operating limit and during normal operation, change in the APLHGR is slow. Therefore, the proposed Frequency is sufficient to assure the parameter remains within limits and the change does not significantly increase the consequences of a previously evaluated accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change introduces no new mode of plant operation nor does it require physical modification to the plant. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change has no impact on any safety analysis assumption since the verification of operation within the APLHGR limit is still required and is consistent with those assumptions. The proposed Surveillance Frequency has been determined through engineering judgement to be adequate for assuring the APLHGR does not exceed the limits. Therefore, the change does not involve a significant reduction in a margin of safety.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The deletion of the Surveillance when operating with a LIMITING CONTROL ROD PATTERN for APLHGR will have minimal effect on the probability or consequences of an accident since operating at the parameter limit does not invalidate safety analysis assumptions. Additionally, it would not be evident that a LIMITING CONTROL ROD PATTERN for APLHGR had been achieved until the 24 hour Frequency Surveillance was performed. As a result, the 24 hour Frequency Surveillance serves to assure the parameter does not exceed the limits. This Frequency has been demonstrated through operating experience to be adequate. Therefore, no significant increase in the probability or consequences of an accident previously evaluated is involved with this change.

2. Does the change 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 require physical modification to the plant. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change has no impact on any safety analysis assumption since operating at the parameter limit is consistent with those assumptions. The existing 24 hour Surveillance Frequency is maintained and has been demonstrated through operating experience to be adequate for assuring the parameter does not exceed limits. Therefore, the change does not involve a significant reduction in a margin of safety.

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The change to the Surveillance Frequency will require the verification of the MCPR limit only once during low power operations with periodic reverification to identify trends. The MCPR limit is used to verify the unit is operating within the initial assumptions of the safety analysis. Significant changes in this parameter are indicative of unanticipated operation, but are not, in themselves, identified as initiators of any previously analyzed accident. Therefore, the change in Frequency of the Surveillance will not significantly increase the probability of an accident previously identified. At low power, there are large inherent margins to the MCPR operating limit and during normal operation, change in the MCPR is slow. Therefore, the proposed Frequency is sufficient to assure the parameter remains within limits and the change does not significantly increase the consequences of a previously evaluated accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change introduces no new mode of plant operation nor does it require physical modification to the plant. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change has no impact on any safety analysis assumption since the verification of operation within the MCPR limit is still required and is consistent with those assumptions. The proposed Surveillance Frequency has been determined through engineering judgement to be adequate for assuring the MCPR does not exceed the limits. Therefore, the change does not involve a significant reduction in a margin of safety.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The deletion of the Surveillance when operating with a LIMITING CONTROL ROD PATTERN for MCPR will have minimal effect on the probability or consequences of an accident since operating at the parameter limit does not invalidate safety analysis assumptions. Additionally, it would not be evident that a LIMITING CONTROL ROD PATTERN for MCPR had been achieved until the 24 hour Frequency Surveillance was performed. As a result, the 24 hour Frequency Surveillance serves to assure the parameter does not exceed the limits. This Frequency has been demonstrated through operating experience to be adequate. Therefore, no significant increase in the probability or consequences of an accident previously evaluated is involved with this change.

2. Does the change 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 require physical modification to the plant. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change has no impact on any safety analysis assumptions since operating at the parameter limit is consistent with those assumptions. The existing 24 hour Surveillance Frequency is maintained and has been demonstrated through operating experience to be adequate for assuring the parameter does not exceed limits. Therefore, the change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.2.3 - LINEAR HEAT GENERATION RATE

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The change to the Surveillance Frequency will require the verification of the LHGR limit only once during low power operations with periodic reverification to identify trends. The LHGR limit is used to verify the unit is operating within the initial assumptions of the safety analysis. Significant changes in this parameter are indicative of unanticipated operation, but are not, in themselves, identified as initiators of any previously analyzed accident. Therefore, the change in Frequency of the Surveillance will not significantly increase the probability of an accident previously identified. At low power, there are large inherent margins to the LHGR operating limit and during normal operation, change in the LHGR is slow. Therefore, the proposed Frequency is sufficient to assure the parameter remains within limits and the change does not significantly increase the consequences of a previously evaluated accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change introduces no new mode of plant operation nor does it require physical modification to the plant. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change has no impact on any safety analysis assumption since the verification of operation within the LHGR limit is still required and is consistent with those assumptions. The proposed Surveillance Frequency has been determined through engineering judgement to be adequate for assuring the LHGR does not exceed the limits. Therefore, the change does not involve a significant reduction in a margin of safety.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The deletion of the Surveillance when operating with a LIMITING CONTROL ROD PATTERN for LHGR will have minimal effect on the probability or consequences of an accident since operating at the parameter limit does not invalidate safety analysis assumptions. Additionally, it would not be evident that a LIMITING CONTROL ROD PATTERN for LHGR had been achieved until the 24 hour Frequency Surveillance was performed. As a result, the 24 hour Frequency Surveillance serves to assure the parameter does not exceed the limits. This Frequency has been demonstrated through operating experience to be adequate. Therefore, no significant increase in the probability or consequences of an accident previously evaluated is involved with this change.

2. Does the change 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 require physical modification to the plant. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change has no impact on any safety analysis assumption since operating at the parameter limit is consistent with those assumptions. The existing 24 hour Surveillance Frequency is maintained and has been demonstrated through operating experience to be adequate for assuring the parameter does not exceed limits. Therefore, the change does not involve a significant reduction in a margin of safety.

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Reactor Power is not considered as an initiator of any analyzed event. In addition, neither the failure to post a notice concerning the APRM gains, nor the APRM gains themselves are considered as an initiator of any analyzed event. While the initial power level is assumed as an initial condition of many accidents, this change will not affect the requirement to maintain power level within the assumptions of the accident analysis. The Quad Cities 1 and 2 Operating Licenses will continue to require Quad Cities 1 and 2 to not exceed 100% of RTP. Therefore, the proposed change does not significantly increase the probability or consequences of a previously evaluated accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change introduces no new mode of plant operation nor does it require physical modification to the plant. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change has no impact on any safety analysis assumption since the requirement to maintain power less than or equal to 100% RTP, as specified in the Operating License, is unchanged. In addition, failure to post a notice that the APRM gains have been adjusted will not increase the potential for exceeding 100% RTP. Therefore, the change does not involve a significant reduction in a margin of safety.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The change to the Surveillance Frequency will require the verification of APRM Gain and Setpoint limits only once during low power operations with periodic reverification to identify trends. The APRM Gain and Setpoint limits is used to verify the unit is operating within the initial assumptions of the safety analysis. Significant changes in this parameter are indicative of unanticipated operation, but are not, in themselves, identified as initiators of any previously analyzed accident. Therefore, the change in Frequency of the Surveillance will not significantly increase the probability of an accident previously identified. At low power, there are large inherent margins to the APLHGR, LHGR, and MCPR operating limits and during normal operation, changes in APLHGR, LHGR, and MCPR are slow. At higher power levels, core peaking is reduced and therefore the need to adjust the APRMs or flow biased scram setpoints is reduced. However, since core nuclear instrumentation is monitored, any anomalies will be detected and corrected between required Surveillances during any power ascension. Therefore, the proposed Frequency is sufficient to assure the parameter remains within limits and the change does not significantly increase the consequences of a previously evaluated accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change introduces no new mode of plant operation nor does it require physical modification to the plant. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change has no impact on any safety analysis assumption since the verification of operation within the FDLRC or MFLPD limit is still required and is consistent with those assumptions. The proposed Surveillance Frequency has been determined through engineering judgement to be adequate for assuring the APRM Gain and Setpoint does not exceed the limits. Therefore, the change does not involve a significant reduction in a margin of safety.

ENVIRONMENTAL ASSESSMENT
ITS: SECTION 3.2 - POWER DISTRIBUTION LIMITS

In accordance with the criteria set forth in 10 CFR 50.21, ComEd has evaluated this proposed Technical Specification change for identification of licensing and regulatory actions requiring environmental assessment, determined it meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9) and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92(b). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or which changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria:

1. The amendment involves no significant hazards consideration.

   As demonstrated in the No Significant Hazards Consideration, this proposed amendment does not involve any significant hazards consideration.

2. There is no significant change in the type or significant increase in the amounts of any effluents that may be released offsite.

   The proposed change will not result in changes in the operation or configuration of the facility. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no change in the types or significant increase in the amounts of any effluents released offsite resulting from this change.

3. There is no significant increase in individual or cumulative occupational radiation exposure.

   The proposed change will not result in changes in the operation or configuration of the facility which impact radiation exposure. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from this change.

Therefore, based upon the above evaluation, ComEd has concluded that no irreversible consequences exist with the proposed change.