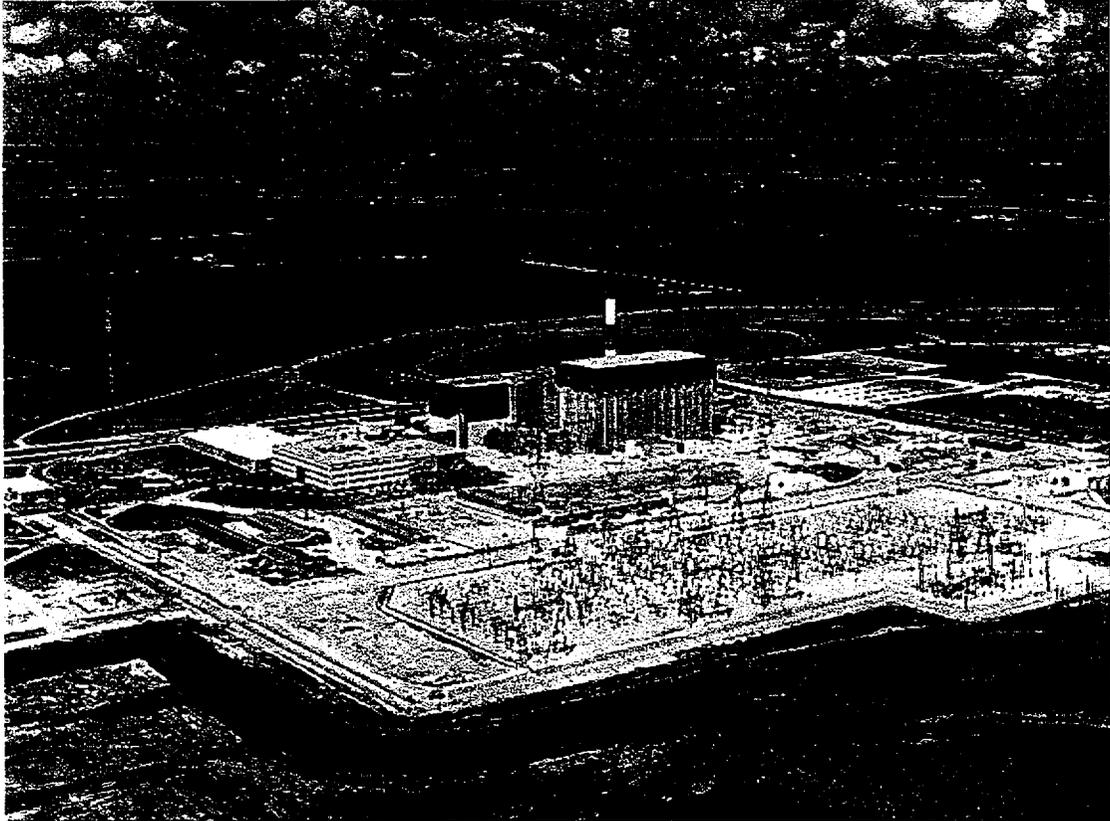


# Improved Technical Specifications



## LaSalle County Station

Volume 2:  
Sections 3.1 and 3.2

**ComEd**

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LC0 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.  <u>AND</u>	Immediately  (continued)

ACTIONS

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. (continued)	E.2 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
	<u>AND</u>	
	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
<p>SR 3.1.1.1 Verify SDM is:</p> <ul style="list-style-type: none"> <li>a. <math>\geq 0.38\% \Delta k/k</math> with the highest worth control rod analytically determined; or</li> <li>b. <math>\geq 0.28\% \Delta k/k</math> with the highest worth control rod determined by test.</li> </ul>	<p>Prior to each in vessel fuel movement during fuel loading sequence</p> <p><u>AND</u></p> <p>Once within 4 hours after criticality following fuel movement within the reactor pressure vessel or control rod replacement</p>

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Reactivity Anomalies

LC0 3.1.2 The reactivity difference between the monitored core  $k_{eff}$  and the predicted core  $k_{eff}$  shall be within  $\pm 1\% \Delta k/k$ .

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 $\pm 1\% \Delta k/k$ .	Once within 24 hours after reaching equilibrium conditions following startup after fuel movement within the reactor pressure vessel or control rod replacement  <u>AND</u>  1000 MWD/T thereafter during operations in MODE 1

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Control Rod OPERABILITY

LC0 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
<p>A. One withdrawn control rod stuck.</p>	<p>-----NOTE----- Rod Worth Minimizer (RWM) may be bypassed as allowed by LC0 3.3.2.1, "Control Rod Block Instrumentation," if required to allow continued operation. -----</p>	
	<p>A.1 Verify stuck control rod separation criteria are met.  <u>AND</u>  A.2 Disarm the associated control rod drive (CRD).  <u>AND</u></p>	<p>Immediately</p> <p>2 hours</p> <p>(continued)</p>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	<p>A.3 Perform SR 3.1.3.2 and SR 3.1.3.3 for each withdrawn OPERABLE control rod.</p> <p><u>AND</u></p> <p>A.4 Perform SR 3.1.1.1.</p>	<p>24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM</p> <p>72 hours</p>
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.	<p>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. -----</p> <p>Fully insert inoperable control rod.</p> <p><u>AND</u></p> <p>C.2 Disarm the associated CRD.</p>	<p>3 hours</p> <p>4 hours</p>

(continued)



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 the 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 notch position 05 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
<p>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 <math>\geq</math> 800 psig.</p>	<p>120 days cumulative operation in MODE 1</p>
<p>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.</p>	<p>Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect scram time</p>
<p>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 <math>\geq</math> 800 psig.</p>	<p>Prior to exceeding 40% RTP after fuel movement within the affected core cell</p> <p><u>AND</u></p> <p>Prior to exceeding 40% RTP after work on control rod or CRD System that could affect scram time</p>

Table 3.1.4-1  
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 05. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."
- 

NOTCH POSITION	SCRAM TIMES <sup>(a)(b)</sup> (seconds) when reactor steam dome pressure $\geq$ 800 psig
45	0.41
39	0.80
25	1.77
05	3.20

- (a) Maximum scram time from fully withdrawn position based on de-energization of scram pilot valve solenoids as 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

LC0 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. -----	8 hours
	Declare the associated control rod scram time "slow."  <u>OR</u> A.2 Declare the associated control rod inoperable.	

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. Two or more control rod scram accumulators inoperable with reactor steam dome pressure <math>\geq</math> 900 psig.</p>	<p>B.1 Restore charging water header pressure to <math>\geq</math> 940 psig.</p> <p><u>AND</u></p> <p>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. -----</p> <p>Declare the associated control rod scram time "slow."</p> <p><u>OR</u></p> <p>B.2.2 Declare the associated control rod inoperable.</p>	<p>20 minutes from discovery of Condition B concurrent with charging water header pressure &lt; 940 psig</p> <p>1 hour</p> <p>1 hour</p>
<p>C. One or more control rod scram accumulators inoperable with reactor steam dome pressure &lt; 900 psig.</p>	<p>C.1 Verify all control rods associated with inoperable accumulators are fully inserted.</p> <p><u>AND</u></p>	<p>Immediately upon discovery of charging water header pressure &lt; 940 psig</p> <p>(continued)</p>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	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
<p>A. One or more OPERABLE control rods not in compliance with the analyzed rod position sequence.</p>	<p>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.</p>	<p>8 hours</p>
	<p><u>OR</u> A.2 Declare associated control rod(s) inoperable.</p>	<p>8 hours</p>

(continued)

ACTIONS

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

SURVEILLANCE REQUIREMENTS

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

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LC0 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

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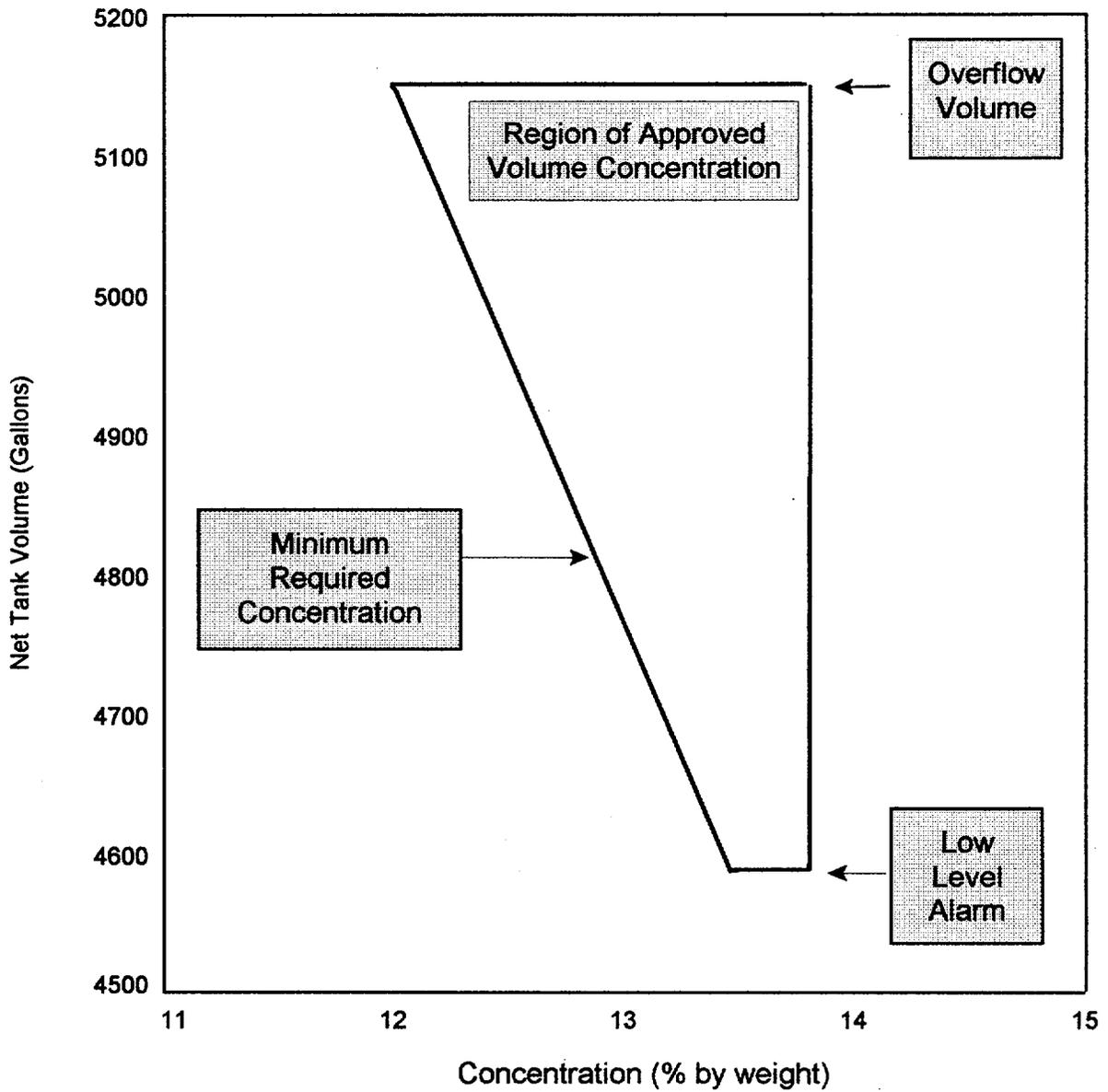
SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
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 up to the storage tank outlet valves is $\geq 68^{\circ}\text{F}$ .	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  <u>AND</u>  Once within 24 hours after water or sodium pentaborate is added to solution  <u>AND</u>  Once within 24 hours after solution temperature is restored within the limits of Figure 3.1.7-2

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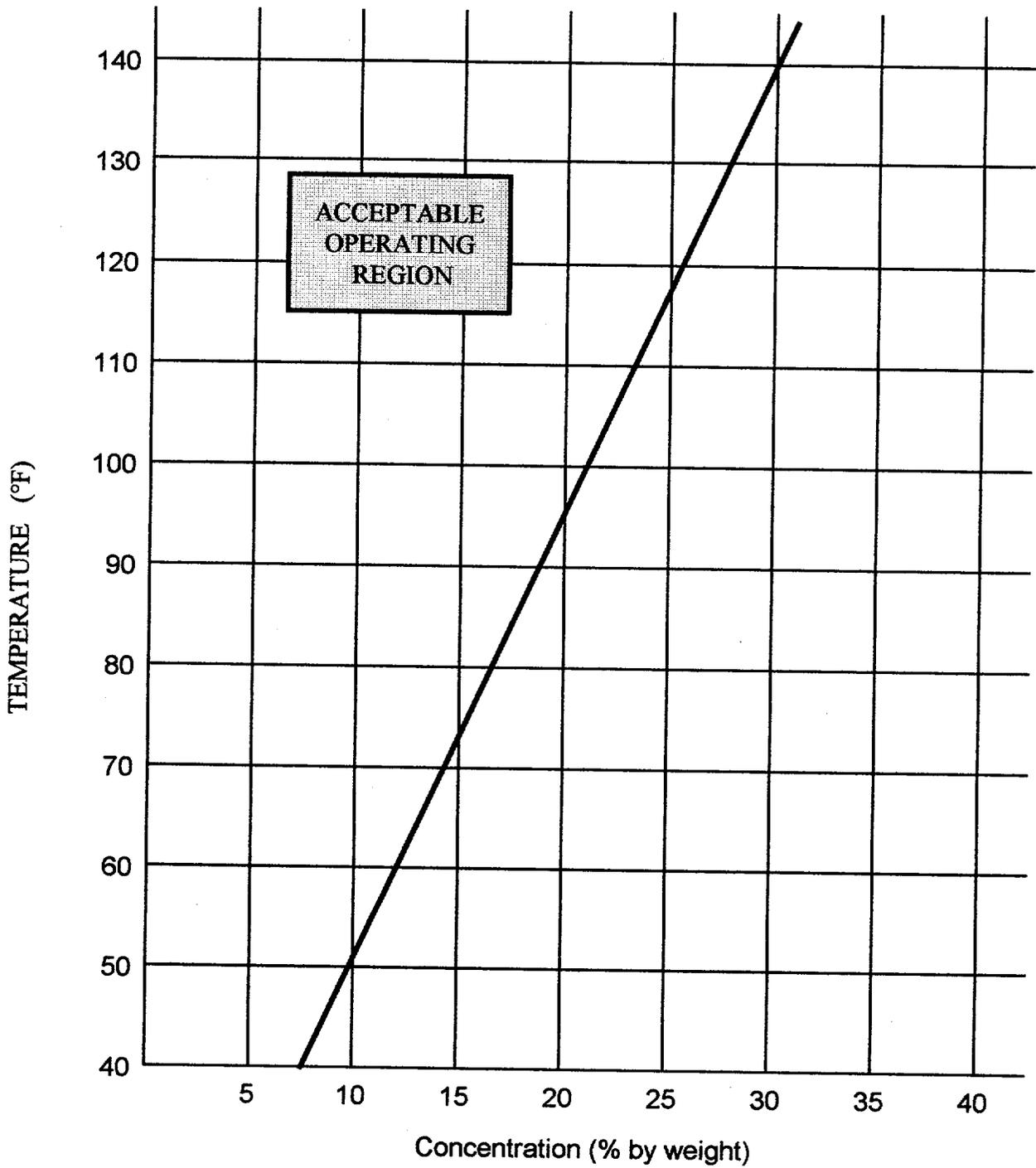
SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.7.6    Verify each SLC subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position, or can be aligned to the correct position.	31 days
SR 3.1.7.7    Verify each pump develops a flow rate $\geq 41.2$ gpm at a discharge pressure $\geq 1220$ 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 storage tank outlet valves 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



**SODIUM PENTABORATE SOLUTION  
VOLUME/CONCENTRATION REQUIREMENTS**

Figure 3.1.7-1 (page 1 of 1)



**SODIUM PENTABORATE SOLUTION  
TEMPERATURE/CONCENTRATION REQUIREMENTS**

Figure 3.1.7-2 (page 1 of 1)

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

LC0 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
<p>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.</p>	<p>31 days</p>
<p>SR 3.1.8.2 Cycle each SDV vent and drain valve to the fully closed and fully open position.</p>	<p>92 days</p>
<p>SR 3.1.8.3 Verify each SDV vent and drain valve:</p> <ul style="list-style-type: none"> <li>a. Closes in <math>\leq 30</math> seconds after receipt of an actual or simulated scram signal; and</li> <li>b. Opens when the actual or simulated scram signal is reset.</li> </ul>	<p>24 months</p>

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

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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 GDC 26 (Ref. 1), which can compensate for the reactivity effects of the fuel and water temperature changes experienced during all operating conditions.

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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 a refueling accident (Ref. 2). 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

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BASES

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APPLICABLE SAFETY ANALYSES 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).

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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).

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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 inadvertent criticality during the withdrawal of a single control rod from a core cell containing one or more fuel assemblies (Ref. 2).

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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 6 hour Completion time is acceptable, considering that the reactor can still be shut down, assuming no additional failures of control rods to insert, and the low probability of an event occurring during this interval.

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BASES

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ACTIONS  
(continued)

B.1

If the SDM cannot be restored, the plant must be brought to MODE 3 within 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. Actions 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 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.

SURVEILLANCE  
REQUIREMENTSSR 3.1.1.1

Adequate SDM must be verified to ensure 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 (i.e., BOC is the most reactive point in the cycle), no correction to the BOC measured value is required (Ref. 4). For the SDM demonstrations that rely solely on calculation of the highest worth control rod, additional margin (0.10%  $\Delta k/k$ ) must be added to the SDM limit of 0.28%  $\Delta k/k$  to account for uncertainties in the calculation.

(continued)

## BASES

SURVEILLANCE  
REQUIREMENTSSR 3.1.1.1 (continued)

The SDM may be demonstrated during an in-sequence control rod withdrawal, in which the highest worth control rod is analytically determined, or during local criticals, where the highest worth control rod is determined by testing. Local critical tests require the withdrawal of out of sequence control rods. This testing would therefore require bypassing of the Rod Worth Minimizer to allow the out of sequence withdrawal, and therefore additional requirements must be met (see LCO 3.10.6, "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 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 also required to ensure 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 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 or reload sequences inherently satisfy the SR, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM.

## REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. UFSAR, Section 15.4.1.1.
3. UFSAR, Section 4.3.2.4.1.
4. NEDE-24011-P-A, "GE 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

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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. 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, 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 ensuring 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

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(continued)

BASES

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BACKGROUND  
(continued)

excess positive reactivity is compensated by burnable 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  $k_{\text{effective}}$  ( $k_{\text{eff}}$ ), 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 monitored  $k_{\text{eff}}$  is calculated by the core monitoring system for actual plant conditions and is then compared to the predicted value for the cycle exposure.

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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  $k_{\text{eff}}$  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  $k_{\text{eff}}$  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  $k_{\text{eff}}$  from the predicted  $k_{\text{eff}}$  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 satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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(continued)

BASES (continued)

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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 core  $k_{eff}$  and the predicted core  $k_{eff}$  of 1%  $\Delta k/k$  has been established based on engineering judgment. A > 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

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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, control rod 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.

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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)

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BASES

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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 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.

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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 further 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

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.1.2.1 (continued)

subsequent replacement with a new control rod or a control rod from another core location. Also, core reactivity changes during the cycle. The 24 hour interval after reaching equilibrium conditions following a startup is based on the need for equilibrium xenon concentrations in the core, such that an accurate comparison between the monitored and predicted core  $k_{eff}$  values 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.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.
  2. UFSAR, Chapter 15.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Control Rod OPERABILITY

BASES

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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 that under conditions of normal operation, including anticipated operational occurrences, specified acceptable fuel design limits are not exceeded. In addition, the control rods provide the capability to hold the reactor core subcritical under all conditions and to limit the potential amount and rate of reactivity increase caused by a malfunction in the CRD System. The CRD System is designed to satisfy the requirements of GDC 26, GDC 27, GDC 28, and GDC 29, (Ref. 1).

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

This Specification, along with LCO 3.1.4, "Control Rod Scram Times," 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, 4, 5, and 6.

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APPLICABLE  
SAFETY ANALYSES

The analytical methods and assumptions used in the evaluations involving control rods are presented in References 2, 3, 4, 5, and 6. 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

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APPLICABLE  
SAFETY ANALYSES  
(continued)

The capability of inserting 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 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 that results 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 (APLGR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), 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. 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).

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LCO

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 and therefore an inoperable accumulator does not

(continued)

BASES

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LCO  
(continued) 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 includes correct assembly of the CRD housing supports.

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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."

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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. The Required Actions are modified by a Note that allows the Rod Worth Minimizer (RWM) to be bypassed if required to allow continued operation. LCO 3.3.2.1, "Control Rod Block Instrumentation," provides additional requirements when the RWM is bypassed to ensure compliance with the CRDA analysis. With one withdrawn control rod stuck, the local scram reactivity rate assumptions may not be met if the stuck

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BASES

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

control rod separation criteria are not met. Therefore, a verification that the separation criteria are met must be performed immediately. The separation criteria are not met if: a) the stuck control rod occupies a location adjacent to two "slow" control rods, b) the stuck control rod occupies a location adjacent to one "slow" control rod, and the one "slow" control rod is also adjacent to another "slow" control rod, or c) 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 within 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 amount of 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 for 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 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.

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BASES

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ACTIONS

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

To allow continued operation with a withdrawn control rod stuck, an evaluation of adequate SDM is also required within 72 hours. Should a DBA or transient require a shutdown, to preserve the single failure criterion an additional control rod would have to be assumed to have failed 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.

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

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(continued)

BASES

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ACTIONS

C.1 and C.2 (continued)

within 3 hours and disarmed (electrically or hydraulically) within 4 hours. Inserting a control rod ensures the shutdown and scram capabilities are not adversely affected. The control rod is disarmed to prevent inadvertent withdrawal during subsequent operations. The control rods can be hydraulically disarmed by closing the drive water and exhaust water isolation valves. Electrically, the control rods can be disarmed by disconnecting power from all four directional control valve solenoids. Required Action C.1 is modified by a Note that allows the RWM to be bypassed if required to allow insertion of the inoperable control rods and continued operation. LCO 3.3.2.1 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  $\leq 10\%$  RTP, the analyzed rod position sequence analysis (Refs. 7 and 8) 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. A Note has been added to the Condition to clarify that the Condition is not applicable when  $> 10\%$  RTP since the analyzed rod position sequence 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.

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(continued)

BASES

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ACTIONS  
(continued)

E.1

If any Required Action and associated Completion Time of Condition A, C, or D are not met or nine or more inoperable control rods exist, 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 (i.e., 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 conditions in an orderly manner and without challenging plant systems.

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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 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.

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BASES

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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.

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(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS  
(continued)

SR 3.1.3.4

Verifying the scram time for each control rod to notch position 05 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) 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 anytime 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.

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BASES (continued)

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- REFERENCES
1. 10 CFR 50, Appendix A, GDC 26, GDC 27, GDC 28, and GDC 29.
  2. UFSAR, Section 4.3.2.5.
  3. UFSAR, Section 4.6.1.1.2.
  4. UFSAR, Section 5.2.2.2.
  5. UFSAR, Section 15.4.
  6. UFSAR, Section 15.4.9.
  7. NEDO-21231, "Banked Position Withdrawal Sequence," Section 7.2, January 1977.
  8. NFSR-0091, Commonwealth Edison Topical Report, Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods, (as specified in Technical Specification 5.6.5).
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Control Rod Scram Times

BASES

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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 scrambled 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 valves 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 accumulator pressure drops 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 within the required time without assistance from reactor pressure.

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APPLICABLE  
SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the control rod scram function are presented in References 2, 3, 4, 5, and 6. 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 scrambling slower than the average time, with several control rods scrambling 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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(continued)

BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

The scram function of the CRD System protects the MCPR Safety Limit (SL) (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), and the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), 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 design limits during reactivity insertion accidents (see Bases for LCO 3.1.6, "Rod Pattern Control"). For the reactor vessel overpressure protection analysis (Ref. 4), 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).

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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. To account for single failure and "slow" scrambling 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 to allow up to 7.0% of the control rods (e.g.,  $185 \times 7.0\% \approx 12$ ) to have scram times that exceed the specified limits (i.e., "slow" control rods) assuming a single stuck control rod (as allowed by LCO 3.1.3, "Control Rod OPERABILITY") and an additional control rod failing to scram per the single failure criterion. The scram times are specified as a function of reactor steam dome pressure to account for the pressure dependence of the scram times. The scram times are specified relative to measurements based on reed switch positions, which provide the control rod position indication. The reed switch closes ("pickup") when the index tube passes a specific location and then opens ("dropout") as the index tube travels upward. Verification of the specified scram times in Table 3.1.4-1 is accomplished through measurement of the "dropout" times.

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BASES

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LCO  
(continued)

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 (face or diagonal) locations.

Table 3.1.4-1 is modified by two Notes, which state 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.

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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."

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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.

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(continued)

BASES (continued)

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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 assumed control rod scram time. Measurement of the scram times with reactor steam dome pressure  $\geq 800$  psig demonstrates acceptable scram times for the transients analyzed in References 5 and 6.

Maximum scram insertion times occur at a reactor 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  $\geq 800$  psig ensures that the 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 scram time testing is performed within a reasonable time following a shutdown  $\geq 120$  days, control rods are required to be tested before exceeding 40% RTP. This Frequency is acceptable, considering the additional Surveillances performed for control rod OPERABILITY, the frequent verification of adequate accumulator pressure, and the required testing of control rods affected by fuel movement within the associated core cell and by work on control rods or the CRD System.

SR 3.1.4.2

Additional testing of a sample of control rods is required to verify the continued performance of the scram function during the cycle. A representative sample contains at least

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.1.4.2 (continued)

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." If more than 20% of the sample is 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 were 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 that the affected control rod is still within acceptable limits. The scram time limits for reactor pressures < 800 psig are found in the Technical Requirements Manual (Ref. 7) and are established based on a high probability of meeting the acceptance criteria at reactor pressures  $\geq$  800 psig. Limits for reactor pressures  $\geq$  800 psig are found in Table 3.1.4-1. If testing demonstrates the affected control rod does not meet these limits, but is within 7-second limit of Table 3.1.4-1, Note 2, the control rod can be declared OPERABLE and "slow."

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(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.1.4.3 (continued)

Specific examples of work that could affect the scram times include (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 valves in the piping required for scram.

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

SR 3.1.4.4

When work that could affect the scram insertion time is performed on a control rod or CRD System, or when fuel movement within the reactor pressure vessel occurs, testing must be done to demonstrate each affected control rod is still within the limits of Table 3.1.4-1 with the reactor steam dome pressure  $\geq$  800 psig. Where work has been performed at high reactor pressure, the requirements of SR 3.1.4.3 and SR 3.1.4.4 will be satisfied with one test. For a control rod affected by work performed while shut down, however, a zero pressure and a high pressure test may be required. This testing ensures that the control rod scram performance is acceptable for operating reactor pressure conditions prior to withdrawing the control rod for continued operation. Alternatively, a 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 of testing the control rod at the different conditions and the more frequent surveillances on other aspects of control rod OPERABILITY.

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(continued)

BASES (continued)

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- REFERENCES
1. 10 CFR 50, Appendix A, GDC 10.
  2. UFSAR, Section 4.3.2.5.
  3. UFSAR, Section 4.6.1.1.2.
  4. UFSAR, Section 5.2.2.
  5. UFSAR, Section 15.4.
  6. UFSAR, Section 15.4.9.
  7. Technical Requirements Manual.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Control Rod Scram Accumulators

BASES

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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."

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APPLICABLE  
SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the control rod scram function are presented in References 1, 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. 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 the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), which ensure that no fuel damage will occur if these limits are not exceeded (see Bases for LCO 3.1.4). Also, 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

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APPLICABLE SAFETY ANALYSES (continued) Control rod scram accumulators satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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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.

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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 under these conditions. Requirements for scram accumulators in MODE 5 are contained in LCO 3.9.5, "Control Rod OPERABILITY - Refueling."

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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 action 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.

A.1 and A.2

With one control rod scram accumulator inoperable and the reactor steam dome pressure  $\geq 900$  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, which clarifies that declaring the control rod "slow" is only applicable if the associated

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BASES

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ACTIONS

A.1 and A.2 (continued)

control rod scram time was 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 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 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 considered 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  $\geq 900$  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 considered a reasonable time to place a CRD pump into service to restore the charging header pressure, if required. This Completion Time also recognizes 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" is only applicable if the associated

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(continued)

BASES

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ACTIONS

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

control rod scram time was 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 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 considered 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 scram 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 the accumulator is inoperable.

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(continued)

BASES

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ACTIONS  
(continued)

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 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 Required 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.

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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 980 psig to 1200 psig. 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.

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REFERENCES

1. UFSAR, Section 4.3.2.5.3.
  2. UFSAR, Section 4.6.1.1.2.
  3. UFSAR, Section 5.2.2.2.2.3.
  4. UFSAR, Section 15.4.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Rod Pattern Control

BASES

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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 effectively 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.

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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 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

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BASES

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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 the 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 defined to minimize the maximum incremental control rod worths 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 analyses may also be performed for atypical operating conditions (e.g., fuel leaker suppression).

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

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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.

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(continued)

BASES (continued)

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APPLICABILITY In MODES 1 and 2, when THERMAL POWER is  $\leq$  10% RTP, the CRDA is a Design Basis Accident (DBA) and, therefore, compliance with the assumptions of the safety analysis is required. When THERMAL POWER is  $>$  10% RTP, there is no credible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel design limit during a CRDA (Ref. 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.

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ACTIONS

A.1 and A.2

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

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

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BASES

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ACTIONS

A.1 and A.2 (continued)

required by Required Action A.2. The allowed Completion Time of 8 hours is reasonable, considering the restrictions on the number of allowed out of sequence control rods and the low probability of a CRDA occurring during the time the control rods are out of sequence.

B.1 and B.2

If nine or more OPERABLE control rods are out of sequence, the control rod pattern significantly deviates from the prescribed sequence. Control rod withdrawal should be suspended immediately to prevent the potential for further deviation from the prescribed sequence. Control rod insertion to correct control rods withdrawn beyond their allowed position is allowed since, in general, insertion of control rods has less impact on control rod worth than withdrawals have. Required Action B.1 is modified by a Note that 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).

With nine or more OPERABLE control rods not in compliance with analyzed rod position sequence, the reactor mode switch must be placed in the shutdown position within 1 hour. With the reactor mode switch in shutdown, the reactor is shut down, and therefore 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.

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BASES (continued)

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

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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, "NRC Safety Evaluation Report for GESSAR II BWR/6 Nuclear Island Design, Docket No. 50-447," Section 4.2.1.3.2, April 1983.
7. NUREG-0800, "Standard Review Plan," Section 15.4.9, "Radiological Consequences of Control Rod Drop Accident (BWR)," Revision 2, July 1981.
8. NEDO-21778-A, "Transient Pressure Rises Affected Fracture Toughness Requirements for Boiling Water Reactors," December 1978.

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BASES

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REFERENCES  
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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, "Determination of Exclusion Area Low Population Zone and Population Center Distance."
  12. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.
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## B 3.1 REACTIVITY CONTROL SYSTEMS

### B 3.1.7 Standby Liquid Control (SLC) System

#### BASES

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#### 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 (ATWS).

The SLC System consists of a boron solution storage tank, two positive displacement pumps, two explosive valves, which 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.

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#### 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 not enough control rods can be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to compensate for all of the various reactivity effects that could occur during plant operation. To meet this objective, it is necessary to inject a quantity of boron that produces a reactivity change equivalent to a concentration of 660 ppm of enriched boron in the reactor core at 68°F. To ensure this objective is met, a sodium pentaborate solution enriched with boron-10 is used. The shutdown analysis assumes a sodium pentaborate solution with enriched boron is used (Ref. 2). A 45% enriched sodium pentaborate solution is also used to satisfy the requirements of Reference 1. To allow for potential leakage and imperfect mixing in the reactor system, an additional amount of boron equal to 25% of the amount cited above is added (Ref. 2). An additional 250 ppm is provided to

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BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

accommodate dilution in the RPV by the residual heat removal shutdown cooling piping. The volume versus concentration limits in Figure 3.1.7-1 are calculated such that the required concentration is achieved. 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.

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

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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 containing an OPERABLE pump, an explosive valve and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path.

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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 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 during these conditions, when only a single control rod can be withdrawn.

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ACTIONS

A.1

If one SLC System 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 perform the shutdown function. However, the overall reliability is reduced because a single failure in

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BASES

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ACTIONS

A.1 (continued)

the remaining OPERABLE subsystem could result in reduced SLC System shutdown capability and inability to meet the requirements of Reference 1. The 7 day Completion Time is based on the availability of an OPERABLE subsystem capable of performing the unit shutdown function and the low probability of a Design Basis Accident (DBA) or severe transient occurring concurrent with the failure of the Control Rod Drive 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.

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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 the SLC System OPERABILITY without disturbing normal plant operation. These Surveillances ensure the proper borated solution and temperature, including the temperature (using the local indicator) of the pump suction piping up to the storage tank outlet valves, are maintained. Maintaining a minimum specified borated solution temperature is important in

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BASES

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SURVEILLANCE  
REQUIREMENTS

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

ensuring that the boron remains in solution and does not precipitate out in the storage tank or in the pump suction piping. The 24 hour Frequency of these SRs is based on operating experience that 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 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 that has demonstrated the reliability of the explosive charge continuity.

SR 3.1.7.6 verifies each valve in the system is in its correct position, but does not apply to the squib (i.e., explosive) valves. Verifying the correct alignment for manual, power operated, and automatic valves in the SLC System flow path ensures 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 does not apply to valves that are locked, sealed, or otherwise secured in position, since they were verified to be in the correct position prior to locking, sealing, or securing. This verification of valve alignment does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct positions. The 31 day Frequency is based on engineering judgment and is consistent with the procedural controls governing valve operation that ensure correct valve positions.

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(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS  
(continued)

SR 3.1.7.5

This Surveillance requires an examination of the sodium pentaborate solution by using chemical analysis to ensure 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 establish that the sodium pentaborate solution concentration is within the specified limits. This Surveillance must be performed anytime the temperature is restored to within the limits of Figure 3.1.7-1, to ensure 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 each SLC System pump develops a flow rate  $\geq 41.2$  gpm at a discharge pressure  $\geq 1220$  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)

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BASES

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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 test 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 in the flow path between the boron solution storage tank and the storage tank outlet valves 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 up to the storage tank outlet valves is unblocked is to verify flow from the storage tank to the test tank. Upon completion of this verification, the pump suction piping between the storage tank outlet valve and pump suction must be drained and flushed with demineralized water, since the piping is not heat traced. 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 daily 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 within the limits of Figure 3.1.7-2.

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REFERENCES

1. 10 CFR 50.62.
  2. UFSAR, Section 9.3.5.3.
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B 3.1 REACTIVITY CONTROL SYSTEMS

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

BASES

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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 consists of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two headers and two instrument volumes, each receiving approximately one half of the control rod drive (CRD) discharges. The two instrument volumes are connected to a common drain line with two valves in series. Each header is connected to a common vent line with 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.

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APPLICABLE  
SAFETY ANALYSES

The Design Basis Accident and transient analyses assume all the control rods are capable of scrambling. The primary function of the SDV is to limit the amount of reactor coolant discharged during a scram. 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 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 also

(continued)

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BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

allow continuous drainage of the SDV during normal plant operation to ensure 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 exceeds a specified setpoint. The setpoint is chosen such 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).

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LCO

The OPERABILITY of all SDV vent and drain valves ensures that, during a scram, the SDV vent and drain valves will close 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 be open to ensure that a path is available for the SDV piping to drain freely at other times.

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APPLICABILITY

In MODES 1 and 2, a scram may be required, and 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.

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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.

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(continued)

BASES

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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 during the time 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

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(continued)

BASES

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ACTIONS

C.1 (continued)

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

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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 function 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 a receipt of a scram signal is based on the bounding leakage case evaluated in the accident analysis. Similarly, after receipt of a simulated or actual scram

(continued)

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BASES

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SURVEILLANCE      SR 3.1.8.3 (continued)

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.

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REFERENCES

1. UFSAR, Section 4.6.1.1.2.
  2. 10 CFR 100.
  3. NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," August 1981.
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3/4.1 REACTIVITY CONTROL SYSTEMS

A.1

3/4.1.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

LC 3.1.1 3.1.1 The SHUTDOWN MARGIN shall be equal to or greater than:

- a. 0.38% delta k/k with the highest worth rod analytically determined, or
- b. 0.28% delta k/k with the highest worth rod determined by test.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

- ACTION A a. In OPERATIONAL CONDITION 1 or 2, reestablish the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.
- ACTION B
- ACTION C b. In OPERATIONAL CONDITION 3 or 4, immediately verify all insertable control rods to be inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL CONDITION 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.
- ACTION D
- ACTION E c. In OPERATIONAL CONDITION 5, suspend CORE ALTERATIONS and other activities that could reduce the SHUTDOWN MARGIN and insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

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

- SR 3.1.1.1 a. By measurement, prior to or during the first startup after each refueling.
  - b. By measurement, within 500 MVDT prior to the core average exposure at which the predicted SHUTDOWN MARGIN, including uncertainties and calculation biases, is equal to the specified limit.
  - c. Within 12 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or is untrippable, except that the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod.
- ← Add proposed first frequency to S.R. 3.1.1.1

A.1

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

LCD 3.1.1

3.1.1 The SHUTDOWN MARGIN shall be equal to or greater than:

- a. 0.38% delta k/k with the highest worth rod analytically determined, or
- b. 0.28% delta k/k with the highest worth rod determined by test.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

ACTION A

- a. In OPERATIONAL CONDITION 1 or 2, reestablish the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION B

ACTIONS C and D

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

ACTION D

ACTION E

- c. In OPERATIONAL CONDITION 5, suspend CORE ALTERATIONS and other activities that could reduce the SHUTDOWN MARGIN and insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

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

SR3.1.1.1

- a. By measurement, prior to or during the first startup after each refueling.
- b. By measurement, within 500 MWD/T prior to the core average exposure at which the predicted SHUTDOWN MARGIN, including uncertainties and calculation biases, is equal to the specified limit.
- c. Within 12 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or is untrippable, except that the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod.

add proposal first frequency to S.R. 3.1.1.1

DISCUSSION OF CHANGES  
ITS: 3.1.1 - SHUTDOWN MARGIN

ADMINISTRATIVE

- A.1 In the conversion of the LaSalle 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-1434, 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 3.1.1 Action b words of "verify...inserted," the ITS 3.1.1 Required Actions C.1 and D.1 are active -- "Initiate action to insert..." This wording provides the same intent in the event all insertable control rods are inserted, and is therefore administrative.
- A.3 CTS 3.1.1 Actions b and c 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 3.1.1 Actions b and c have been deleted.
- A.4 The CTS 3.1.1 Actions b and c 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 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.

DISCUSSION OF CHANGES  
ITS: 3.1.1 - SHUTDOWN MARGIN

ADMINISTRATIVE

A.4 (cont'd) This interpretation of the Actions intent is supported by the BWR ISTS, NUREG-1434, 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 3.1.1 Actions b and c 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-1434, Revision 1, and is considered administrative only.

A.6 Not used.

A.7 The CTS 3.1.1 Action c to "insert...within 1 hour" (see Discussion of Change L.2 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 comment A.4. 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-1434, Revision 1. Because this is an enhanced presentation of existing intent, the proposed change is considered administrative.

DISCUSSION OF CHANGES  
ITS: 3.1.1 - SHUTDOWN MARGIN

ADMINISTRATIVE (continued)

- A.8        A specific completion time for the SDM test required by CTS 4.1.1.a is proposed to clarify when "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. 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-1434, Revision 1. Because this is an enhanced presentation of existing intent, the proposed change is considered administrative.
- A.9        More explicit wording is proposed to replace the activity referred to as "refueling" in CTS 4.1.1.a. 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.10       The CTS 4.1.1.c 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-1434, 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

- M.1        An additional Surveillance Frequency for SDM verification is proposed to be added to CTS 4.1.1.a (proposed SR 3.1.1.1, first Frequency) to clarify the requirements necessary for assuring SDM during the refueling process. Because SDM is assumed in several refueling mode analyses in the UFSAR, assurance that intermediate fuel loading patterns have adequate SDM is necessary. This change imposes a requirement where none is explicitly provided in the existing Technical Specifications. As discussed in the Bases corresponding to this Surveillance Frequency, the SDM verification in these situations is best accomplished by analysis (rather than in-sequence criticals) because of the many changes in core loading during a typical refueling. Bounding analyses may be used to demonstrate adequate SDM for the most reactive configurations during refueling, thereby showing acceptability of the entire fuel movement sequence.

DISCUSSION OF CHANGES  
ITS: 3.1.1 - SHUTDOWN MARGIN

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

None

"Specific"

- L.1        The CTS 3.1.1 Action c requirement to suspend all CORE ALTERATIONS 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.7. 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 3.1.1 Action c 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.
- L.3        CTS 4.1.1.b has been deleted. The SDM limits adequately account for uncertainties and biases, and for fuel cycle changes. As long as the required margin is met, as determined by the initial startup test and corroborated by the periodic reactivity anomaly Surveillance (CTS 4.1.2, proposed SR 3.1.2.1), there should be no need for additional Surveillance Requirements. Furthermore, the requirement to perform an additional demonstration just prior to when the predicted SDM equals the limit, would require a shutdown of the plant. This shutdown to perform the test would be in addition to the shutdown that is

DISCUSSION OF CHANGES  
ITS: 3.1.1 - SHUTDOWN MARGIN

TECHNICAL CHANGES - LESS RESTRICTIVE

L.3 required a short time later when the SDM requirement is no longer met. The  
(cont'd) acceptability of the proposed Surveillances, which do not include an additional  
test just prior to the SDM limit not being met, is further supported in the BWR  
ISTS, NUREG-1434, Revision 1 presentation of these requirements.

RELOCATED SPECIFICATIONS

None

REACTIVITY CONTROL SYSTEM

A.1

3/4.1.2 REACTIVITY ANOMALIES

LIMITING CONDITION FOR OPERATION

LCO  
3.1.2

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% delta k/k.

A.2

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

Core k<sub>eff</sub>

M.1

ACTION:

With the reactivity different by more than 1% delta k/k:

L.1

ACTION  
A

a. Within <sup>(72)</sup> 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.

LA.1

ACTION  
B

b. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

SR  
3.1.2.1

4.1.2 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% delta k/k:

A.2

a. During the first startup following CORE ALTERATIONS, and

Core k<sub>eff</sub>

M.1

b. At least once per 37 effective full power days during POWER OPERATION.

A.3

L.2

1000 MWD/T

L.3

A.1

REACTIVITY CONTROL SYSTEM

3/4.1.2 REACTIVITY ANOMALIES

LIMITING CONDITION FOR OPERATION

LC03.1.2

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% delta k/k.

A.2

Core Keff

M.1

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the reactivity different by more than 1% delta k/k:

L.1

ACTION A

a. Within <sup>72</sup> 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.A.1

ACTION B

b. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

A.2

SR

3.1.2.1

4.1.2 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% delta k/k:

M.1

a. During the first startup following CORE ALTERATIONS, and

A.3

b. At least once per 31 effective full power days during POWER OPERATION.

L.2

L.3

Core Keff

M.1

1000 MW D/T

DISCUSSION OF CHANGES  
ITS: 3.1.2 - REACTIVITY ANOMALIES

ADMINISTRATIVE

- A.1 In the conversion of the LaSalle 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-1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).
- A.2 The wording "reactivity equivalence of the difference" in CTS 3.1.2 and CTS 4.1.2 has been changed to "reactivity difference" to be consistent with NUREG-1434, 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 4.1.2.a 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 to 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-1434, Revision 1. Because this is an enhanced presentation of existing intent, the proposed change is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M.1 CTS 3/4.1.2 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  $k_{\text{eff}}$  with the predicted  $k_{\text{eff}}$  as calculated by an approved 3-D core simulator code. These two methods to meet CTS 3/4.1.2

DISCUSSION OF CHANGES  
ITS: 3.1.2 - REACTIVITY ANOMALIES

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 (cont'd) were previously approved by the NRC in the SER for Amendment Nos. 116 and 101, dated October 29, 1996. Since LaSalle 1 and 2 predicts the core reactivity using a 3-D simulator code and compares 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 3.1.2 Action a 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 with the limit is maintained in Required Action A.1. As a result, these details associated with the method of restoring compliance with 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 CTS 3.1.2 Action a (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

DISCUSSION OF CHANGES  
ITS: 3.1.2 - REACTIVITY ANOMALIES

TECHNICAL CHANGES - LESS RESTRICTIVE

- L.1 (cont'd) considerations, the BWR ISTS, NUREG-1434, Revision 1 allows this time to be extended to 72 hours.
- L.2 The term "CORE ALTERATIONS" in CTS 4.1.2.a 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 4.1.2.b "31 effective full power days" (approximately 734 MWD/T) is proposed to be replaced with "1000 MWD/T." 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-1434, Revision 1.

RELOCATED SPECIFICATIONS

None

A.1

REACTIVITY CONTROL SYSTEM

3/4.1.3 CONTROL RODS

CONTROL ROD OPERABILITY

LIMITING CONDITION FOR OPERATION

<general reorganization> A.2

LC03.1.3 3.1.3.1 All control rods shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

add proposed ACTIONS NOTE

add proposed Required Action A.1 Note

ACTION A

a. With one control rod inoperable due to being immovable, as a result of excessive friction or mechanical interference, or known to be untrippable.

1. Within 1 hour:

(ACTION D)

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

b) Disarm the associated directional control valves\* either:

1) Electrically, or

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

c) Comply with Surveillance Requirement 4.1.1.c.

2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

3. Restore the inoperable control rod to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION C

b. With one or more control rods trippable but inoperable for causes other than addressed in ACTION a, above:

1. If the inoperable control rod(s) is withdrawn:

a) Immediately verify:

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

2) The insertion capability of the inoperable withdrawn control rod(s) by inserting the control rod(s) at least one notch by drive water pressure within the normal operating range\*\*.

b) Otherwise, insert the inoperable withdrawn control rod(s) and disarm the associated directional control valves\* either.

1) Electrically, or

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

add proposed Required Action C.1 Note

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

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

A.1

ITS 3.1.3

REACTIVITY CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

ACTION C 2. If the inoperable control rod(s) is inserted: L.4

a) Within <sup>(4)</sup>hour disarm the associated directional control valves either:

- 1) Electrically, or
- 2) Hydraulically by closing the drive water and exhaust water isolation valves.

LRD  
LA.1

ACTION E b) Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

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

- d\*. With one or more SDV vent or drain lines with one valve inoperable,
  - 1. Isolate\*\* the associated line within 7 days.
  - 2. Otherwise, be in HOT SHUTDOWN within the next 12 hours.
- e\*. With one or more SDV vent or drain lines with both valves inoperable,
  - 1. Isolate\*\* the associated line within 8 hours.
  - 2. Otherwise, be in HOT SHUTDOWN within the next 12 hours.

A.7  
moved to  
LC03.18

SURVEILLANCE REQUIREMENTS

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

- a. At least once per 31 days verifying each valve to be open\*\*, and
- b. At least once per 92 days cycling each valve through at least one complete cycle of full travel

A.7  
moved to  
LC03.18

SR3.1.3.2 4.1.3.1.2 When above the low power setpoint of the RWM, all withdrawn control rods ~~not required to have their directional control valves disarmed electrically~~  
 SR3.1.3.3 ~~or hydraulically~~ shall be demonstrated OPERABLE by moving each control rod at least one notch:

A.8

- a. At least once per 7 days, and
- b. ~~At least once per~~ 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference.

M.5  
L.5  
L.6

Required  
Action A.3

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

A.6

These valves may be closed intermittently for testing under administrative control.  
 \*Separate Action statement entry is allowed for each SDV vent and drain line.  
 \*\*An isolated line may be unisolated under administrative control to allow draining and venting of the SDV.

A.7  
moved to  
LC03.18

A.1

REACTIVITY CONTROL SYSTEMSURVEILLANCE REQUIREMENTS (Continued)

4.1.3.1.3 All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.1.3.2, 4.1.3.4, 4.1.3.5, 4.1.3.6 and 4.1.3.7.

A.9

4.1.3.1.4 The scram discharge volume shall be determined OPERABLE by demonstrating the scram discharge volume drain and vent valves OPERABLE at least once per 18 months by verifying that the drain and vent valves:

A.7  
moved  
to  
LCO 3.1.8

- a. Close within 30 seconds after receipt of a signal for control rods to scram, and
- b. Open after the scram signal is reset.

A.1

3/4.1.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1 The SHUTDOWN MARGIN shall be equal to or greater than:

- a. 0.38% delta k/k with the highest worth rod analytically determined, or
- b. 0.28% delta k/k with the highest worth rod determined by test.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

- a. In OPERATIONAL CONDITION 1 or 2, reestablish the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.
- b. In OPERATIONAL CONDITION 3 or 4, immediately verify all insertable control rods to be inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL CONDITION 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.
- c. In OPERATIONAL CONDITION 5, suspend CORE ALTERATIONS and other activities that could reduce the SHUTDOWN MARGIN, and insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

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

Required Action A.4

a. By measurement, prior to or during the first startup after each refueling.

b. By measurement, within 500 MWD/T prior to the core average exposure at which the predicted SHUTDOWN MARGIN, including uncertainties and calculation biases, is equal to the specified limit.

See ITS 3.1.1

Required Action A.4

c. Within 12 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or is untrippable, except that the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod.

L.7  
A.5  
A.10

Moved to ITS Chapter 1.0

**REACTIVITY CONTROL SYSTEM**

**CONTROL ROD MAXIMUM SCRAM INSERTION TIMES**

**LIMITING CONDITION FOR OPERATION**

*(general organization)*

A.11

SR 3.1.3.4

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position 05, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7.0 seconds.

A.12

**APPLICABILITY:** OPERATIONAL CONDITIONS 1 and 2.

**ACTION:**

*ACTION A & C*

With the maximum scram insertion time of one or more control rods exceeding 7.0 seconds:

1. Declare the control rod(s) with the slow insertion time inoperable, and

2. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of 7.0 seconds.

L.6

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

*add proposed SR 3.1.3.4*

A.13

**SURVEILLANCE REQUIREMENTS**

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

- a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS\* or after a reactor shutdown that is greater than 120 days,
- b. For specifically affected individual control rods 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
- c. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of operation.

*(See ITS 3.1.4)*

\*Except normal control rod movement.

REACTIVITY CONTROL SYSTEM

CONTROL ROD DRIVE COUPLING

LIMITING CONDITION FOR OPERATION

SR3.1.3.5 3.1.3.6 All control rods shall be coupled to their drive mechanisms A.14

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 5\* L.9

ACTION:

ACTION C a. In OPERATIONAL CONDITIONS 1 and 2 with one control rod not coupled to its associated drive mechanism: L.4

1. Within 2 hours, either: L.10

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.15

1) Observing any indicated response of the nuclear instrumentation, and L.11

2) Demonstrating that the control rod will not go to the overtravel position. L.12

ACTION C b) If recoupling is not accomplished on the first attempt or, if not permitted by the RWM then until permitted by the RWM, declare the control rod inoperable and insert the control rod and disarm the associated directional control valves either: L.10

- 1) Electrically, or
- 2) Hydraulically by closing the drive water and exhaust water isolation valves. CRD  
LA.1

ACTION E 2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5\* with a withdrawn control rod not coupled to its associated drive mechanism, within 2 hours, either: L.9

1. Insert the control rod to accomplish recoupling and verify recoupling by withdrawing the control rod and demonstrating that the control rod will not go to the overtravel position, or
2. If recoupling is not accomplished, insert the control rod and disarm the associated directional control valves\*\* either:
  - a) Electrically, or
  - b) Hydraulically by closing the drive water and exhaust water isolation valves.

\*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2 L.9

\*\*May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status. A.6

A.1

REACTIVITY CONTROL SYSTEMSURVEILLANCE REQUIREMENTS

SR 3.1.3.5 4.1.3.6 A control rod shall be demonstrated to be coupled to its drive mechanism by observing any indicated response of the nuclear instrumentation while withdrawing the control rod to the fully withdrawn position and then verifying that the control rod drive does not go to the overtravel position:

L.12

- a. Prior to reactor criticality after completing CORE ALTERATIONS that could have affected the control rod drive coupling integrity,
- b. Anytime the control rod is withdrawn to the "Full out" position in subsequent operation, and
- c. Following maintenance on or modification to the control rod or control rod drive system which could have affected the control rod drive coupling integrity.

A.16

A.1

ITS 3.1.3

REACTIVITY CONTROL SYSTEM

CONTROL ROD POSITION INDICATION

LIMITING CONDITION FOR OPERATION

SR 3.1.3.1 3.1.3.7 The control rod position indication system shall be OPERABLE

A.17

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5\*

A.18

moved to ITS 3.9.4

ACTION:

**ACTION C** a. In OPERATIONAL CONDITION 1 or 2 with one or more control rod position indicators inoperable, within one hour:

1. Determine the position of the control rod by:

- (a) Moving the control rod, by single notch movement, to a position with an OPERABLE position indicator,
- (b) Returning the control rod, by single notch movement, to its original position, and

LA.2

(c) Verifying no control rod drift alarm at least once per 12 hours, or

L.13

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

LA.2

3. When THERMAL POWER is:

Required Action C.1 Note

(a) Within the low power setpoint of the RWM:

1) Declare the control rod inoperable.

M.6

2) Verify the position and bypassing of control rods with inoperable "Full in" and/or "Full out" position indicators by a second licensed operator or other technically qualified member of the unit technical staff.

A.19

(b) Greater than the low power setpoint of the RWM, declare the control rod inoperable, insert the control rod and disarm the associated directional control valves\*\* either:

1) Electrically, or

CRD

LA.11

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

**ACTION E** 4. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

\*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

A.18

moved to ITS 3.9.4

\*\*May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE STATUS.

A.6

A.1

ITS 3.1.3

REACTIVITY CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

- b. In OPERATIONAL CONDITION 5<sup>o</sup> with a withdrawn control rod position indicator inoperable, move the control rod to a position with an OPERABLE position indicator or insert the control rod.

A.18  
moved to  
ITS 3.9.4

SURVEILLANCE REQUIREMENTS

SR3.1.3.1 4.1.3.7 The control rod position indication system shall be determined OPERABLE by verifying:

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

- b. That the indicated control rod position changes during the movement of the control rod drive when performing Surveillance Requirement 4.1.3.1.2, and
- c. That the control rod position indicator corresponds to the control rod position indicated by the "Full out" position indicator when performing Surveillance Requirement 4.1.3.6b.
- d. That the control rod position indicator corresponds to the control rod position indicated by the "Full in" position indicator:
  1. Prior to each reactor startup, and
  2. Each time a control rod is fully inserted.

L.14

\*At least each withdrawn control rod not applicable to control rods removed per Specifications 3.9.10.1 or 3.9.10.2.

A.18  
moved to  
ITS 3.9.4

A.1

REACTIVITY CONTROL SYSTEM  
3/4.1.3 CONTROL RODS  
CONTROL ROD OPERABILITY  
LIMITING CONDITION FOR OPERATION

<general reorganization> A.2

LC03.1.3 3.1.3.1 All control rods shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, and 2.

ACTION:

add proposed ACTION 5 NOTE

add propose & Require & Action A.1 Note

A.3

A.4

A.5

ACTION A

a. With one control rod inoperable due to being immovable, as a result of excessive friction or mechanical interference, or known to be untrippable:

1. Within 1 hour:

(ACTION D) a) Verify that the inoperable control rod, if withdrawn, is separated from all other inoperable control rods by at least two control cells in all directions.

L.1

M.1

M.2

b) Disarm the associated directional control valves\* either:  
1) Electrically, or  
2) Hydraulically by closing the drive water and exhaust water isolation valves.

Control rod drive (CRD)

Required Actions A.2 and A.4

L.2

c) Comply with Surveillance Requirement 4.1.1.c.

add & proposed Required Action A.1

LA.1

ACTION E.2.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

L.3

3. Restore the inoperable control rod to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

M.2

ACTION C

b. With one or more control rods trippable but inoperable for causes other than addressed in ACTION a, above:

1. If the inoperable control rod(s) is withdrawn:

a) Immediately verify:

(ACTION D) 1) That the inoperable withdrawn control rod(s) is separated from all other inoperable withdrawn control rod(s) by at least two control cells in all directions, and

add proposed ACTION B

L.4

2) The insertion capability of the inoperable withdrawn control rod(s) by inserting the control rod(s) at least one notch by drive water pressure within the normal operating range\*\*.

M.3

Required Actions C.1 and C.2

b) Otherwise, insert the inoperable withdrawn control rod(s) and disarm the associated directional control valves\* either:

1) Electrically, or  
2) Hydraulically by closing the drive water and exhaust water isolation valves

CRD  
LA.1

add proposed Required Action C.1 Note

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

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

A.4

A.6

M.3

A.1

REACTIVITY CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

ACTION C

2. (If the inoperable control rod(s) is inserted:

L.4

a) Within 1 hour disarm the associated directional control valves either:

CRD

LA.1

1) Electrically, or

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

ACTION E

b) Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

ACTION E

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

d\*. With one or more SDV vent or drain lines with one valve inoperable,

A.7

moved to ITS 3.1.8

- 1. Isolate\*\* the associated line within 7 days.
- 2. Otherwise, be in HOT SHUTDOWN within the next 12 hours.

e\*. With one or more SDV vent or drain lines with both valves inoperable,

- 1. Isolate\*\* the associated line within 8 hours.
- 2. Otherwise, be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

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

A.7

moved to ITS 3.1.8

- a. At least once per 31 days verifying each valve to be open\*\*, and
- b. At least once per 92 days cycling each valve through at least one complete cycle of full travel.

SR3.1.3.2

4.1.3.1.2 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.8

SR3.1.3.3

a. At least once per 7 days, and

M.S

L.5

Required Action A.3

b. At least once per 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference.

L.6

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

A.6

\*\*These valves may be closed intermittently for testing under administrative control.

A.7

\*\*\*Separate Action statement entry is allowed for each SDV vent and drain line.

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

moved to ITS 3.1.8

A.1

ITS 3.1.3

REACTIVITY CONTROL SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

~~4.1.3.1.3 All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.1.3.2; 4.1.3.4, 4.1.3.5, 4.1.3.6 and 4.1.3.7.~~

A.9

4.1.3.1.4 The scram discharge volume shall be determined OPERABLE by demonstrating the scram discharge volume drain and vent valves OPERABLE at least once per 18 months by verifying that the drain and vent valves:

A.7

moved to  
LC3.1.8

- a. Close within 30 seconds after receipt of a signal for control rods to scram, and
- b. Open after the scram signal is reset.

3/4.1 REACTIVITY CONTROL SYSTEMS

A.1

3/4.1.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1 The SHUTDOWN MARGIN shall be equal to or greater than:

- a. 0.38% delta k/k with the highest worth rod analytically determined, or
- b. 0.28% delta k/k with the highest worth rod determined by test.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

- a. In OPERATIONAL CONDITION 1 or 2, reestablish the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.
- b. In OPERATIONAL CONDITION 3 or 4, immediately verify all insertable control rods to be inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL CONDITION 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.
- c. In OPERATIONAL CONDITION 5, suspend CORE ALTERATIONS and other activities that could reduce the SHUTDOWN MARGIN, and insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

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

<See ITS 3.1.1>

- a. By measurement, prior to or during the first startup after each refueling.
- b. By measurement, within 500 MWD/T prior to the core average exposure at which the predicted SHUTDOWN MARGIN, including uncertainties and calculation biases, is equal to the specified limit.

- c. Within <sup>72</sup>2 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or is untrippable, except that the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod.

L.7

A.5

A.10

(Mined to ITS)  
Chapter 1.0  
Amendment No. 121

Required Action A.4

Required Action A.4

REACTIVITY CONTROL SYSTEM

A.1

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

<general organization>

A.11

SR 3.1.3.4

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position 05, ~~based on de-energization of the scram pilot valve solenoids as time zero~~ shall not exceed 7.0 seconds.

A.12

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

ACTION A or C

With the maximum scram insertion time of one or more control rods exceeding 7.0 seconds:

1. Declare the control rod(s) with the slow insertion time inoperable, and

2. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of 7.0 seconds.

L.8

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

add proposed SR 3.1.3.4

A.13

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

- a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS or after a reactor shutdown that is greater than 120 days,
- b. For specifically affected individual control rods 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
- c. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of operation.

See ITS 3.1.4

Except normal control rod movement

A.1

REACTIVITY CONTROL SYSTEM

CONTROL ROD DRIVE COUPLING

LIMITING CONDITION FOR OPERATION

SR3.1.3.5 3.1.3.6 All control rods shall be coupled to their drive mechanisms. A.14

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 5\*. L.9

ACTION:

ACTION C

a. In OPERATIONAL CONDITIONS 1 and 2 with one control rod not coupled to its associated drive mechanism: L.4

1. Within 2 hours, either: L.10

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.15 L.11

1) Observing any indicated response of the nuclear instrumentation, and L.12

2) Demonstrating that the control rod will not go to the overtravel position. L.11

ACTION C

b) If recoupling is not accomplished on the first attempt or, if not permitted by the RWM then until permitted by the RWM, declare the control rod inoperable and insert the control rod and disarm the associated directional control valves either: L.10

1) Electrically, or

2) Hydraulically by closing the drive water and exhaust water isolation valves. CAD L.A.1

ACTION E 2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours. L.9

b. In OPERATIONAL CONDITION 5\* with a withdrawn control rod not coupled to its associated drive mechanism, within 2 hours, either:  
1. Insert the control rod to accomplish recoupling and verify recoupling by withdrawing the control rod and demonstrating that the control rod will not go to the overtravel position, or  
2. If recoupling is not accomplished, insert the control rod and disarm the associated directional control valves\*\* either:  
a) Electrically, or  
b) Hydraulically by closing the drive water and exhaust water isolation valves.

\*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2. L.9

\*\*May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status. A.6

A.1

REACTIVITY CONTROL SYSTEMSURVEILLANCE REQUIREMENTS

SR3.1.3.5 4.1.3.6 A control rod shall be demonstrated to be coupled to its drive mechanism by observing any indicated response of the nuclear instrumentation while withdrawing the control rod to the fully withdrawn position and then verifying that the control rod drive does not go to the overtravel position:

L.12

- a. Prior to reactor criticality after completing CORE ALTERATIONS that could have affected the control rod drive coupling integrity,
- b. Anytime the control rod is withdrawn to the "Full out" position in subsequent operation, and
- c. Following maintenance on or modification to the control rod or control rod drive system which could have affected the control rod drive coupling integrity.

A.16

A.1

ITS 3.1.3

REACTIVITY CONTROL SYSTEM

CONTROL ROD POSITION INDICATION

LIMITING CONDITION FOR OPERATION

SR 3.1.3.1 3.1.3.7 The control rod position indication system shall be OPERABLE.

A.17

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 (and 5\*)

A.18

ACTION:

MOVED to ITS 3.9.4

**ACTION C** a. In OPERATIONAL CONDITION 1 or 2 with one or more control rod position indicators inoperable within one hour:

1. Determine the position of the control rod by:

(a) Moving the control rod, by single notch movement, to a position with an OPERABLE position indicator,

LA.2

(b) Returning the control rod, by single notch movement, to its original position, and

(c) Verifying no control rod drift alarm at least once per 12 hours, or

L.13

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

LA.2

3. When THERMAL POWER is:

(a) Within the low power setpoint of the RWM:

Required Action C.1 Note

(1) Declare the control rod inoperable,

M.6

(2) Verify the position and bypassing of control rod with inoperable "Full in" and/or "Full out" position indicators by a second licensed operator or other technically qualified member of the unit technical staff.

A.19

b) Greater than the low power setpoint of the RWM, declare the control rod inoperable, insert the control rod and disarm the associated directional control valves\*\* either:

(1) Electrically, or

CRD

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

LA.1

**ACTION E** 4. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

\*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

A.18

MOVED to ITS 3.9.4

\*\*May be reamed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

A.6

A.1

ITS 3.1.3

REACTIVITY CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

b. In OPERATIONAL CONDITION 5<sup>\*</sup> with a withdrawn control rod position indicator inoperable, move the control rod to a position with an OPERABLE position indicator or insert the control rod.

A.18  
moved to  
ITS  
3.9.4

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1 4.1.3.7 The control rod position indication system shall be determined OPERABLE by verifying:

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

b. That the indicated control rod position changes during the movement of the control rod drive when performing Surveillance Requirement 4.1.3.1.2, and  
c. That the control rod position indicator corresponds to the control rod position indicated by the "Full out" position indicator when performing Surveillance Requirement 4.1.3.6b.  
d. That the control rod position indicator corresponds to the control rod position indicated by the "Full in" position indicator:  
1. Prior to each reactor startup, and  
2. Each time a control rod is fully inserted.

L.14

\*At least each withdrawn control rod not applicable to control rods removed per Specifications 3.9.10.1 or 3.9.10.2.

A.18  
moved to  
ITS 3.9.4

DISCUSSION OF CHANGES  
ITS: 3.1.3 - CONTROL ROD OPERABILITY

ADMINISTRATIVE

- A.1 In the conversion of the LaSalle 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-1434, 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 sequence, 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 3.1.3.1 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 3.1.3.1, Actions a and b (ITS 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 time, provided the proper ACTIONS of CTS 3.1.4.1 (ITS 3.3.2.1), the RWM Specification, are taken. This is a human factors consideration to assure clarity of the requirement and allowance.

DISCUSSION OF CHANGES  
ITS: 3.1.3 - CONTROL ROD OPERABILITY

ADMINISTRATIVE (continued)

- A.5 The existing phrase of "being immovable, as a result of excessive friction or mechanical interference, or known to be untrippable" in CTS 3.1.3.1 Action a 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 similar phrase in CTS 4.1.1.c has also been changed to "stuck."
- A.6 CTS 3.1.3.1 Actions a.1.b), b.1.b), and b.2.a), footnote \*, CTS 3.1.3.6 Action a.1.b) footnote \*\*, and CTS 3.1.3.7 Action a.3.b) footnote \*\*, 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.7 The SDV vent and drain valves requirements in CTS 3.1.3.1 Actions d and e, CTS 4.1.3.1.1, and CTS 4.1.3.1.4 are being moved to ITS 3.1.8 in accordance with the format of the BWR ISTS, NUREG-1434, Revision 1. Any technical changes to these requirements will be discussed in the Discussion of Changes for ITS: 3.1.8.
- A.8 CTS 4.1.3.1.2 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 4.0.3), and therefore, CTS 4.1.3.1.2 only applies to OPERABLE control rods. Therefore, this phrase is proposed to be deleted since it is not needed.
- A.9 These listed Surveillances in CTS 4.1.3.1.3 are required by other Specifications. Repeating a requirement to perform these Surveillances is not necessary. Elimination of this "cross-reference" is therefore administrative.
- A.10 The SDM allowance in CTS 4.1.1.c is being moved to the definition of SDM in proposed Section 1.1, in accordance with the BWR ISTS, NUREG-1434, Revision 1. Any technical changes to these requirements will be discussed in the Discussion of Changes for ITS Chapter 1.0.
- A.11 The CTS 3.1.3.2 requirement that maximum control rod scram insertion time be  $\leq 7$  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

DISCUSSION OF CHANGES  
ITS: 3.1.3 - CONTROL ROD OPERABILITY

ADMINISTRATIVE

- A.11 (cont'd) a new or different treatment of the requirements (other than those proposed in Discussion of Change L.8 below). Therefore, this proposed change is administrative.
- A.12 The definition of time zero in CTS 3.1.3.2 (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 3.1.3.3 and 3.1.3.4, 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.13 CTS 4.1.3.2, 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.14 The CTS 3.1.3.6 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.4, L.9, L.10, L.11, and L.12 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.15 CTS 3.1.3.6 Action a.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-1434, 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.16 CTS 4.1.3.6.c addresses the requirement to perform coupling checks after performing activities which could have affected coupling integrity. This Surveillance must be completed prior to allowing the control rod to be considered OPERABLE. (The consideration of OPERABILITY is more clearly presented in the proposed editorial rewrite of CTS 4.1.3.6.c into the Frequency for proposed SR 3.1.3.5.) Therefore, CTS 4.1.3.6.a is redundant. "CORE ALTERATIONS that could have affected the control rod drive coupling integrity" is a subset the CTS 4.1.3.6.c requirement, "maintenance...which

DISCUSSION OF CHANGES  
ITS: 3.1.3 - CONTROL ROD OPERABILITY

ADMINISTRATIVE

A.16 (cont'd) could have affected the control rod drive coupling integrity." Performance of the integrity verification prior to control rod OPERABILITY (which is the understanding of CTS 4.1.3.6.c as presented in the proposed SR 3.1.3.5) bounds "prior to reactor criticality." Therefore, elimination of CTS 4.1.3.6.a is administrative and represents no change in requirements.

A.17 The intent of the CTS 3.1.3.7 requirement is understood to be related to each control rod. The Applicability footnote "\*", each specific action within Action a, Action b, 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 (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 3.1.3.7 Actions a.1 and a.2 address this intent. The proposed SR 3.1.3.1 has combined the CTS 3.1.3.7 intent with the CTS 3.1.3.7 Actions a.1 and a.2 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 3.1.3.7 Action a.1 or a.2, or meeting proposed SR 3.1.3.1.

A.18 The CTS 3.1.3.7 requirements, including Action b, for control rod position indication during refueling (OPERATIONAL CONDITION 5) are being moved to Section 3.9 in accordance with the format of the BWR ISTS, NUREG-1434, Revision 1. Any technical changes to these requirements will be discussed in the Discussion of Changes for ITS: 3.9.4.

A.19 The requirements of CTS 3.1.3.7 Action a.3.(a)2) are now covered by the Note to ITS 3.1.3 Required Action C.1, which states, in part, that RWM may be bypassed as allowed by proposed LCO 3.3.2.1. LCO 3.3.2.1 provides the requirements of CTS 3.1.3.7 Action a.3.(a)2). Therefore, an explicit ACTION in this Specification to verify the position and bypassing of RWM is not needed.

DISCUSSION OF CHANGES  
ITS: 3.1.3 - CONTROL ROD OPERABILITY

TECHNICAL CHANGES - MORE RESTRICTIVE

- M.1 CTS 3.1.3.1 Actions a.1.a) and b.1.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.
- M.2 The CTS 3.1.3.1 Actions require 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 3.1.3.1 Action a.1.b) 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.3 The proposed changes to CTS 3.1.3.1 Action b.1.a)2) including footnote \*\*, 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 3.1.3.1 Action b.1.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.
- M.4 Not used.

DISCUSSION OF CHANGES  
ITS: 3.1.3 - CONTROL ROD OPERABILITY

TECHNICAL CHANGES - MORE RESTRICTIVE (continued)

- 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 4.1.3.1.2 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 CTS 3.1.3.7 Action a.3.(a)1) requires a control rod to be declared inoperable when THERMAL POWER is within the low power setpoint and one or more control rod position indicators is inoperable and control rod position is unknown. CTS 3.1.3.1 Action b for inoperable rods provides the option to verify the insertion capability, and then allows the control rod to remain withdrawn. The proposed changes to the ACTIONS for non-stuck inoperable control rods (ITS 3.1.3 ACTION C) eliminates the check of insertion capability; replacing it with a requirement to fully insert and disarm all inoperable control rods. The effect on the CTS 3.1.3.7 Action a.3.(a)1) for control rods with position unknown, when below the low power setpoint, is to eliminate the option to leave the control rod withdrawn and continue to operate. The control rod will be required to be inserted and disarmed, regardless of the power level (which is currently the requirement if power is greater than the low power setpoint). 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.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA.1 The details of the recommended procedures for disarming control rod drives (CRDs) specified in CTS 3.1.3.1 Actions a.1.b) (with the exception of electrical disarming, see Discussion of Change M.2 above), b.1.b), and b.2.a), CTS 3.1.3.6 Action a.1.b), and CTS 3.1.3.7 Action a.3.(b) 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.

DISCUSSION OF CHANGES  
ITS: 3.1.3 - CONTROL ROD OPERABILITY

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

LA.2 CTS 3.1.3.7 Actions a.1 and a.2, which determine the position of the control rod (now proposed to be a Surveillance for control rod OPERABILITY - refer to Discussion of Change A.17 above) can be met a number of ways. Three ways are presented: by moving the control rod, by single notch movement, to a position with an OPERABLE position indicator, then returning it, by single notch movement, to its original position and periodically verifying no control rod drift alarm, 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 (with the exception of the control rod drift alarm check, see Discussion of Change L.13 below) 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 3.1.3.1 Actions a.1 and b.1.a)1) 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. (Refer to Discussion of Change M.3 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 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.

DISCUSSION OF CHANGES  
ITS: 3.1.3 - CONTROL ROD OPERABILITY

TECHNICAL CHANGES - LESS RESTRICTIVE

- L.1 (cont'd)
- 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 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 3.1.3.2, CTS 3.1.3.3, and 3.1.3.4 (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 3.1.3.1 Action a.1 allows 1 hour and Action b.1.a) requires immediate action. 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.
- L.2 Disarming a control rod as required by CTS 3.1.3.1 Action a.1.b) 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-1434) in recognition of the potential for excessive haste required to

DISCUSSION OF CHANGES  
ITS: 3.1.3 - CONTROL ROD OPERABILITY

TECHNICAL CHANGES - LESS RESTRICTIVE

- L.2 (cont'd) 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 CTS 3.1.3.1 Action a.3, which requires restoration of a stuck control rod within 48 hours, is being deleted. In addition, a new Required Action is being added (ITS 3.1.3 Required Action A.1), since the proposed Specification now allows continued operation with a stuck control rod. With a single withdrawn control rod stuck, the remaining OPERABLE control rods are capable of providing the required scram and shutdown reactivity. During a transient, a single stuck control rod in addition to an assumed single failure will have no significant impact on the established operating limits. SHUTDOWN MARGIN must still be met, accounting for the loss of negative reactivity due to the stuck control rod (refer to the proposed definition of SDM and ITS 3.1.3 Required Action A.4). The stuck rod must also meet certain separation criteria (ITS 3.1.3 Required Action A.1). Prompt action is required to confirm no additional stuck control rods exist (ITS 3.1.3 Required Action A.3). Therefore, continued operation is proposed to be allowed, as are MODE changes in accordance with SR 3.0.4.
- L.4 All inoperable non-stuck control rods are required to be fully inserted and disarmed (refer to Discussion of Change M.3 above). The time allowed to complete the insertion is proposed to be extended to 3 hours for all cases. In the existing ACTIONS for an uncoupled control rod (CTS 3.1.3.6 Action a.1.b)), time is provided to recouple and, if unsuccessful, insert the control rod before entering CTS 3.1.3.1 Action b.1. Two hours are currently allowed to perform these ACTIONS. CTS 3.1.3.1 Action b.1.b) provides no additional time to disarm the control rod (total of 2 hours to insert and an immediate time to disarm). Uncoupled control rod actions are proposed to be addressed by ITS 3.1.3 ACTION C, as are other non-stuck inoperable control rods. This existing 3 hour allowance, before requiring an inoperable (uncoupled) control rod to be inserted, is the time found in the ITS 3.1.3 Required Action C.1 for control rod insertion. For consistency of presentation, this 3 hour limitation is also proposed for all other instances of inoperable control rods. These other instances (loss of position indication, excessive scram speed, certain combinations of conditions with a low pressure on a control rod scram accumulator) also warrant a minimal time to attempt restoration prior to inserting and disarming. It is for these other instances that the extended time to insert are proposed. Since these instances do not represent loss of SDM, and are limited to a total of no more than 8 inoperable control rods, the extended time does not represent a significant safety concern.

DISCUSSION OF CHANGES  
ITS: 3.1.3 - CONTROL ROD OPERABILITY

TECHNICAL CHANGES - LESS RESTRICTIVE

- L.4 (cont'd) 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. 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. 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-1434, 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 is already in its required position (in accordance with other actions), and the action to disarm is solely a mechanism for precluding the potential for future misoperation.
- L.5 CTS 4.1.3.1.2.a, which verifies control rods to be non-stuck, is proposed to be extended from 7 days to 31 days for control rods that are not fully withdrawn (proposed SR 3.1.3.3). This is acceptable given the following:
- 1) At full power, a large percentage of control rods (typically 80% to 90%) are fully withdrawn and would continue to be exercised each week. This represents a significant sample size when looking for an unexpected random event (i.e., a stuck control rod).
  - 2) Operating experience has shown "stuck" control rods to be an extremely rare event while operating.
  - 3) Should a stuck rod be discovered, 100% of the remaining control rods (even partially withdrawn) must be tested within 24 hours (ITS 3.1.3 Required Action A.3).
- L.6 CTS 4.1.3.1.2.b requires a daily notch test in the event power operation is continuing with an immovable control rod and the plant is operating at greater than the low power setpoint of the rod worth minimizer. The ITS requires the control rod notch test only once within 24 hours after the plant is operating at greater than the low power setpoint of the rod worth minimizer (ITS 3.1.3 Required Action A.3). The purpose of the control rod notch test on each withdrawn OPERABLE control rod is to ensure that a generic problem does not exist and that control rod insertion capability remains. The single performance of the control rod notch test satisfies the same function as the daily notch test of the CTS without requiring the additional testing.

DISCUSSION OF CHANGES  
ITS: 3.1.3 - CONTROL ROD OPERABILITY

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

- L.7 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 4.1.1.c has been extended from 12 hours to 72 hours, and provides a reasonable time to perform the analysis or test.
- L.8 The CTS 3.1.3.2 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 ("Otherwise, be in at least HOT SHUTDOWN within 12 hours") could have only applied to CTS 3.1.3.2 Action 2 (since a control rod can always be declared inoperable), this part of the CTS 3.1.3.2 Action has also been deleted.
- L.9 Coupling requirements during refueling (OPERATIONAL CONDITION 5) specified by CTS 3/4.1.3.6 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.10 If an uncoupled control rod is not allowed by the RWM to be inserted to accomplish recoupling, CTS 3.1.3.6 Action a.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

DISCUSSION OF CHANGES  
ITS: 3.1.3 - CONTROL ROD OPERABILITY

TECHNICAL CHANGES - LESS RESTRICTIVE

- L.10 (cont'd) established within 3 hours, the control rod must be fully inserted and disarmed (ITS 3.1.3 Required Actions C.1 and C.2). Also, because of the limited time allowed to recouple, the number of attempts (currently limited to one by CTS 3.1.3.6 Action a.1.b) does not need to be restricted. The number of attempts to recouple a control rod may be restricted by plant procedures, which consider the potential for equipment damage during successive recoupling attempts.
- L.11 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.12 below. As a result, CTS 3.1.3.6 Action a.1.a.2) requirements are proposed to be deleted since they are not necessary for ensuring recoupling of the control rod.
- L.12 The CTS 3.1.3.6 Action a.1.a)1) and CTS 4.1.3.6 requirements 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 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.13 CTS 3.1.3.7 Action a.1 provides methods for determining the position of a control rod whose position indicator is inoperable. These methods require determining position of the control rod by moving the control rod, by single notch movement, to a position with an OPERABLE position indicator, then returning the control rod, by single notch movement, to its original position, and verifying no rod drift alarm is annunciated every 12 hours. The 12 hour

DISCUSSION OF CHANGES  
ITS: 3.1.3 - CONTROL ROD OPERABILITY

TECHNICAL CHANGES - LESS RESTRICTIVE

- L.13 (cont'd) requirement to verify no rod drift alarm is being deleted. The ITS will require the rod position to be determined every 24 hours (proposed SR 3.1.3.1). Thus, if the method of CTS 3.1.3.7 Actions a.1.(a) and (b) is being used to determine the position of a control rod, it will have to be performed every 24 hours. Currently, it has to be performed only once, then the rod drift alarm is used to verify the rod has not moved. In addition, the alarm provides annunciation in the control room and will alarm if any control rod moves; the alarm is not associated with any one single control rod. The probability of a control rod with an inoperable indicator moving is no different than the probability of a control rod with Operable indicators moving. There are numerous controls/indicators available, that would make a mispositioned control rod readily apparent to the operator, such that appropriate actions could be taken, even without the verification of the alarm. This deletion is also consistent with NUREG-1434, Rev. 1, which, while not listing this specific method in the Bases of SR 3.1.3.1, does specify that other appropriate methods to determine rod position can be used (the NUREG allows the utility to determine other appropriate methods, of which CTS 3.1.3.7 Actions a.1.(a) and (b) could be one).
- L.14 Current Surveillance Requirements for the control rod position indication system (CTS 4.1.3.7.b, 4.1.3.7.c, and 4.1.3.7.d) require that the control rod position indication system be determined OPERABLE during the performance of the control rod movement tests (CTS 4.1.3.1.2) and the control rod coupling verifications (CTS 4.1.3.6.b) prior to startup, and each time a control rod is fully inserted. To perform control rod movement tests required by CTS 4.1.3.1.a (proposed SR 3.1.3.2 and SR 3.1.3.3) and control rod coupling verifications required by CTS 4.1.3.6.b (proposed SR 3.1.3.5), position indication must be available. If position indication is not available, these tests 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, SR 3.1.3.3, and SR 3.1.3.5 and are proposed to be deleted.

RELOCATED SPECIFICATIONS

None

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

A.1

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position 05, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7.0 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding 7.0 seconds:

1. Declare the control rod(s) with the slow insertion time inoperable, and
2. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of 7.0 seconds.

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

See ITS 3.1.3

SURVEILLANCE REQUIREMENTS

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

M.1

a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS or after a reactor shutdown that is greater than 120 days,

b. For specifically affected individual control rods 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

M.2

c. For at least 10% of the control rods on a rotating basis at least once per 120 days of operation.

M.2

L.A.1

SR 3.1.4.1  
SR 3.1.4.2  
SR 3.1.4.4

Note to Surveillance Requirements

SR 3.1.4.4  
SR 3.1.4.1  
SR 3.1.4.4  
SR 3.1.4.2

BDD

prior to exceeding 40% RTP

add proposed SR 3.1.4.3

SR 3.1.4.4 Except normal control rod movement

L.1

A.1

ITS 3.1.4

REACTIVITY CONTROL SYSTEM

CONTROL ROD AVERAGE SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.3 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:

add proposed LCO 3.1.4 and Table 3.1.4-1

M.3

Footnote(s) to Table 3.1.4-1

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
45	0.43
39	0.86
25	1.93
05	3.49

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.3.3 All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.1.3.2.

SR 3.1.4.1, SR 3.1.4.2, and SR 3.1.4.4

A.1

REACTIVITY CONTROL SYSTEM

FOUR CONTROL ROD GROUP SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

add proposed LC 3.1.4 and Table 3.1.4-1

3.1.3.4 The average scram insertion time, from the fully withdrawn position, for the three fastest control rods in each group of four control rods arranged in a two-by-two array, (based on deenergization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

Footnote (a) to Table 3.1.4-1

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
45	0.45
39	0.92
25	2.05
05	3.70

M.3

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

1. Declare the control rods with the slower than 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. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with an average scram insertion time(s) in excess of the average scram insertion time limit.

M.3

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.4 All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.1.3.2.

SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.4

REACTIVITY CONTROL SYSTEM

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

A.1

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position 05, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7.0 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding 7.0 seconds:

1. Declare the control rod(s) with the slow insertion time inoperable, and
2. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of 7.0 seconds.

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

See ITS 3.1.3

SURVEILLANCE REQUIREMENTS

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER (following CORE ALTERATIONS) or after a reactor shutdown that is greater than 120 days, M.1

b. For specifically affected individual control rods 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 M.2

c. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of operation. M.2

LA.1

SR 3.1.4.1  
SR 3.1.4.2  
SR 3.1.4.4

SR 3.1.4.1  
SR 3.1.4.1  
SR 3.1.4.1  
SR 3.1.4.2

SR 3.1.4.4 Except normal control rod movement. L.1

REACTIVITY CONTROL SYSTEM  
CONTROL ROD AVERAGE SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.3 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:

add proposed LCo 3.1.4 and Table 3.1.4-1

M.3

Footnote (c) to Table 3.1.4-1

<0531>

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (seconds)
45	0.43
39	0.86
25	1.93
05	3.49

APPLICABILITY: OPERATIONAL CONDITIONS 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.

SURVEILLANCE REQUIREMENTS

4.1.3.3 All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.1.3.2.

SR 3.1.4.1, SR 3.1.4.2, and SR 3.1.4.4

A.1

REACTIVITY CONTROL SYSTEM

FOUR CONTROL ROD GROUP SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

add proposed LCO 3.1.4 and Table 3.1.4-1

3.1.3.4 The average scram insertion time, from the fully withdrawn position, for the three fastest control rods in each group of four control rods arranged in a two-by-two array based on deenergization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

M.3

Footnote (a) to Table 3.1.4-1

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
45	0.45
39	0.92
25	2.05
05	3.70

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

ACTION A

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

M.3

1. Declare the control rods with the slower than 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. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with an average scram insertion time(s) in excess of the average scram insertion time limit.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.4 All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.1.3.2.

SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.4

DISCUSSION OF CHANGES  
ITS: 3.1.4 - CONTROL ROD SCRAM TIMES

ADMINISTRATIVE

- A.1 In the conversion of the LaSalle 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-1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

TECHNICAL CHANGES - MORE RESTRICTIVE

- M.1 The pressure at which the control rods must be tested in CTS 4.1.3.2 (proposed SRs 3.1.4.1, 3.1.4.2, and 3.1.4.4) has been changed from  $\geq 950$  to  $\geq 800$  psig. This pressure corresponds to the limiting pressure for CRD scram testing for the LaSalle 1 and 2 CRD System. "Limiting" refers to the maximum scram times experienced at or below this pressure because of the competing effects of the reactor vessel pressure and the accumulator pressure scram forces. The scram time requirements are related to transients analyzed at rated reactor pressure (assumed to be  $\geq 950$  psig); however, if the scram times are demonstrated at pressures above 800 psig, the measured times are conservative with respect to the conditions assumed in the design basis transient and accident analyses.
- M.2 In the CTS 4.1.3.2.b Surveillance Requirement "for specifically affected" control rods, deleting the flexibility provided in CTS 4.1.3.2 to delay post-maintenance testing until reactor pressure is  $\geq 950$  psig is proposed, to ensure adequate testing is performed prior to declaring the control rod operable (which could include prior to entering MODE 2). In support of the proposed restriction, 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. 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. Furthermore, the CTS 4.1.3.2.b scram time test requirement (proposed SR 3.1.4.4) performed at normal operating reactor pressure, is additionally required to be performed prior to exceeding 40% RTP. This places a finite time on the

DISCUSSION OF CHANGES  
ITS: 3.1.4 - CONTROL ROD SCRAM TIMES

TECHNICAL CHANGES - MORE RESTRICTIVE

M.2 (cont'd) test if maintenance was performed on the control rod in MODE 1 or 2, and ensures the control rod scram times are within the analyzed limits prior to full power operation). It is noted that if the control rod remains inoperable (which requires it to be inserted and disarmed) until normal operating pressures, a single scram time test will satisfy both Surveillance Requirements (i.e., SR 3.1.4.3 and SR 3.1.4.4).

M.3 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 3.1.3.2, 3.1.3.3, and 3.1.3.4 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).

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

DISCUSSION OF CHANGES  
ITS: 3.1.4 - CONTROL ROD SCRAM TIMES

TECHNICAL CHANGES - MORE RESTRICTIVE

M.3 (cont'd) 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 scrambling 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 3.1.3.2 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 4.1.3.2.c 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.

DISCUSSION OF CHANGES  
ITS: 3.1.4 - CONTROL ROD SCRAM TIMES

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

"Specific"

- L.1 CTS 4.1.3.2.a requires control rod scram time testing for all control rods prior to exceeding 40% RTP following CORE ALTERATIONS, except for normal control rod movement (footnote \*). 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 (except for normal control rod movement) 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. 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

REACTIVITY CONTROL SYSTEM  
CONTROL ROD SCRAM ACCUMULATORS  
LIMITING CONDITION FOR OPERATION

LO3.1.5 3.1.3.5 All control rod scram accumulators shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5\*

ACTION:

add proposed Actions Note

A.2  
moved to  
ITS 3.9.5 and  
ITS 3.10.7

ACTION A a. In OPERATIONAL CONDITION 1 or 2:

A.3

1. With one control rod scram accumulator inoperable:

M.1  
and reactor  
Pressure 2900psig

a) Within 8 hours, either:

1) Restore the inoperable accumulator to OPERABLE status, or

A.4

2) Declare the control rod associated with the inoperable accumulator inoperable.

add proposed Required Action A.1

L.1

b) Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

A.5

ACTIONS B and C 2.

With more than one control rod scram accumulator inoperable, declare the associated control rod inoperable and:

M.1

within 1 hour

L.1

a) If the control rod associated with any inoperable scram accumulator is withdrawn, immediately verify that at least one CRD pump is operating by inserting at least one withdrawn control rod at least one notch by drive water pressure within the normal operating range or

add proposed Required Action B.1

L.2

A.6

Required Action  
B.1 Note

place the reactor mode switch in the Shutdown position.

b) Insert the inoperable control rods and disarm the associated directional control valves either:  
1) Electrically, or  
2) Hydraulically by closing the drive water and exhaust water isolation valves.

A.7

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

b. In OPERATIONAL CONDITION 5 with:

1. One withdrawn control rod with its associated scram accumulator inoperable, insert the affected control rod and disarm the associated directional control valves within 1 hour, either:

A.2  
moved to  
ITS 3.9.5 and  
ITS 3.10.7

a) Electrically, or

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

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

\*At least the accumulator associated with each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

A.1

REACTIVITY CONTROL SYSTEMSURVEILLANCE REQUIREMENTS

SR3.1.5.1 4.1.3.5 Each control rod scram accumulator shall be determined OPERABLE:

- a. At least once per 7 days by verifying that the indicated pressure is greater than or equal to 940 psig, unless the control rod is inserted and disarmed or scrambled.

A.8

A.1

ITS 3.1.5

REACTIVITY CONTROL SYSTEM

CONTROL ROD SCRAM ACCUMULATORS

LIMITING CONDITION FOR OPERATION

LC 3.1.5 3.1.3.5 All control rod scram accumulators shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5\*

ACTION:

a. In OPERATIONAL CONDITION 1 or 2:

ACTION A 1. With one control rod scram accumulator inoperable:

a) Within 8 hours, either:

1) Restore the inoperable accumulator to OPERABLE status or

2) Declare the control rod associated with the inoperable accumulator inoperable.

b) Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

ACTIONS B and C 2. With more than one control rod scram accumulator inoperable, declare the associated control rod inoperable and:

a) (If the control rod associated with any inoperable scram accumulator is withdrawn, immediately verify that at least one CRD pump is operating by inserting at least one withdrawn control rod at least one notch by drive water pressure within the normal operating range or place the reactor mode switch in the Shutdown position.

b) Insert the inoperable control rods and disarm the associated directional control valves either:  
1) Electrically, or  
2) Hydraulically by closing the drive water and exhaust water isolation valves.

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

b. In OPERATIONAL CONDITION 5 with:

1. One withdrawn control rod with its associated scram accumulator inoperable, insert the affected control rod and disarm the associated directional control valves within 1 hour, either:

a) Electrically, or

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

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

\*At least the accumulator associated with each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

A.2 moved to ITS 3.9.5 and ITS 3.10.7

add proposed ACTIONS Note

A.3 and reactor pressure 2900psig

A.4 add proposed Required Action A.1

M.1

L.1

A.5

M.1

L.1

Required Action D.1 Note

L.2

A.6

A.7

A.2 moved to ITS 3.9.5 and ITS 3.10.7

A.1

ITS 3.1.5

REACTIVITY CONTROL SYSTEM

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each control rod scram accumulator shall be determined OPERABLE:

SR3.1.5.1

- a. At least once per 7 days by verifying that the indicated pressure is greater than or equal to 940 psig, unless the control rod is inserted and disarmed or scrammed.

AB

DISCUSSION OF CHANGES  
ITS: 3.1.5 - CONTROL ROD SCRAM ACCUMULATORS

ADMINISTRATIVE

- A.1 In the conversion of the LaSalle 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-1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).
- A.2 The CTS 3.1.3.5 requirements, including Action b, for control rod scram accumulator OPERABILITY in MODE 5 are being moved to Sections 3.9 and 3.10 in accordance with the format of the BWR ISTS, NUREG-1434, Revision 1. Any technical changes to these requirements will be discussed in the Discussion of Changes for ITS 3.9.5 and 3.10.7.
- A.3 A proposed ACTIONS Note, "Separate Condition entry is allowed for each control rod scram accumulator," has been added to CTS 3.1.3.5 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 3.1.3.5 Action a.1.a)1) (based on the BWR ISTS, NUREG-1434) 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 ("Otherwise, be in at least HOT SHUTDOWN within the next 12 hours") for failure to perform the CTS 3.1.3.5 Action a.1.a)2) 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 (CTS Action a.1.b)) is inconsequential and considered administrative.

DISCUSSION OF CHANGES  
ITS: 3.1.5 - CONTROL ROD SCRAM ACCUMULATORS

ADMINISTRATIVE (continued)

- A.6 The CTS 3.1.3.5 Action a.2.a) 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 by drive water pressure within the normal operating range 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.7 CTS 3.1.3.5 Action a.2 requires the affected control rod to be declared inoperable. Once declared inoperable, the CTS 3.1.3.1 Actions for an inoperable control rod are required to be taken. The CTS 3.1.3.1 and ITS 3.1.3 ACTIONS for an inoperable control rod contain the requirements to insert and disarm, as well as a shutdown requirement if the Actions are not performed. 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 3.1.3.5 Action a.2.b) has been deleted. Since this change is a presentation preference only, it is considered administrative.
- A.8 These conditions of CTS 4.1.3.5.a, 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 4.0.3 and proposed LCO 3.0.1. Therefore, the stated exception has been deleted.

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 3.1.3.5 Action a.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 ITS 3.1.3 and ITS 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

DISCUSSION OF CHANGES  
ITS: 3.1.5 - CONTROL ROD SCRAM ACCUMULATORS

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 pressures. ITS 3.1.5 ACTION C applies to one or more inoperable  
(cont'd) 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 3.1.3.5 Action a.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 3.1.3.5 Action a.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 3.1.3.5 Action a.2 requirement to declare the associated control rod inoperable immediately. The requirement to declare the associated control rod

DISCUSSION OF CHANGES  
ITS: 3.1.5 - CONTROL ROD SCRAM ACCUMULATORS

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 (cont'd) inoperable (CTS 3.1.3.5 Action a.2) 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 inoperability in CTS 3.1.3.5 Action a.2. This provides a reasonable time to attempt investigation 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 3.1.3.5 Action a.2.a) 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 present ACTIONS more appropriate to the actual plant conditions (in one instance, proposing more restrictive ACTIONS - refer to Discussion of Change M.1 above).

CTS 3.1.3.5 Action a.2.a) 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, such that 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 3.1.3.5 Action a.2.a) 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

DISCUSSION OF CHANGES  
ITS: 3.1.5 - CONTROL ROD SCRAM ACCUMULATORS

TECHNICAL CHANGES - LESS RESTRICTIVE

L.2 operating CRD pump. Restart of this pump or of the spare CRD pump would  
(cont'd) restore charging water 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 water 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

INSERT NEW SPECIFICATION 3.1.6

M.1

Insert new Specification 3.1.6, "Rod Pattern Control," as shown in ITS 3.1.6.

(Unit 1)

Page 1 of 2

ITS 3.1.6

INSERT NEW SPECIFICATION 3.1.6

M.1

Insert new Specification 3.1.6, "Rod Pattern Control," as shown in ITS 3.1.6.

(Unit 2)

Page 2 of 2

DISCUSSION OF CHANGES  
ITS: 3.1.6 - ROD PATTERN CONTROL

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-1434, 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

REACTIVITY CONTROL SYSTEM

A.1

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.1.7 3.1.5 The standby liquid control system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 5\* L.1

ACTION:

a. In OPERATIONAL CONDITION 1 or 2:

- ACTION A 1. With one motor operated suction valve, one pump and/or one explosive valve inoperable, restore the inoperable suction valve, pump and/or explosive valve to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.
- ACTION C
- ACTION B 2. With the standby liquid control system inoperable, restore the system to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.
- ACTION C

b. In OPERATIONAL CONDITION 5\*:

- 1. With one motor operated suction valve, one pump and/or one explosive valve inoperable, restore the inoperable suction valve, pump and/or explosive valve to OPERABLE status within 30 days or insert all insertable control rods within the next hour. L.1
- 2. With the standby liquid control system inoperable, insert all insertable control rods within 1 hour.

SUREILLANCE REQUIREMENTS

4.1.5 The standby liquid control system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that;

- SR 3.1.7.1 1. The available volume and temperature of the sodium pentaborate solution are within the limits of Figures 3.1.5-1 and 3.1.5-2, and
- SR 3.1.7.2 2. The heat tracing circuit is OPERABLE by verifying the indicated temperature ~~to be~~  $\geq 60^{\circ}\text{F}$  on the local indicator. L.A.1
- SR 3.1.7.3 of the pump suction piping up to the storage tank outlet valves and 68

with any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2. L.1 M.1

REACTIVITY CONTROL SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days by;

1. Starting both pumps and recirculating demineralized water to the test tank. L. 2

SR 3.1.7.4

2. Verifying the continuity of the explosive charge.

SR 3.1.7.5

3. Determining that the concentration of boron in solution is within the limits of Figure 3.1.5-2 by chemical analysis.\*

SR 3.1.7.6

4. Verifying that each valve in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

c. At least once per 18 months during shutdown by;

SR 3.1.7.8

LA.2

1. Initiating one of the standby liquid control system loops, including an explosive valve and verifying that a flow path from the pumps to the reactor pressure vessel is available (by pumping demineralized water into the reactor vessel). The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of that batch successfully fired. Both injection loops shall be tested in 48 months. L. 3 A.3

Definition of STAGGERED TEST BASIS

SR 3.1.7.7

2. Demonstrating that when tested pursuant to Specification 4.0.5, the minimum flow requirement of 41.2 gpm at a pressure of greater than or equal to 1220 psig is met. L.D.1

3. Demonstrating that the pump relief valve setpoint is less than or equal to 1400 psig and verifying that the relief valve does not actuate during recirculation to the test tank. LA.4

SR 3.1.7.9

A.4

Storage tank outlet valves are

4. ~~\*\*~~ Demonstrating that all heat traced piping between the storage tank and the reactor vessel is unblocked by verifying flow from the storage tank to the motor operated suction valve and then draining and flushing the piping with demineralized water. A.5 LA.3

Storage tank outlet

5. Demonstrating that the storage tank heaters are OPERABLE by verifying the expected temperature rise for the sodium pentaborate solution in the storage tank after the heaters are energized. L.4 A.8

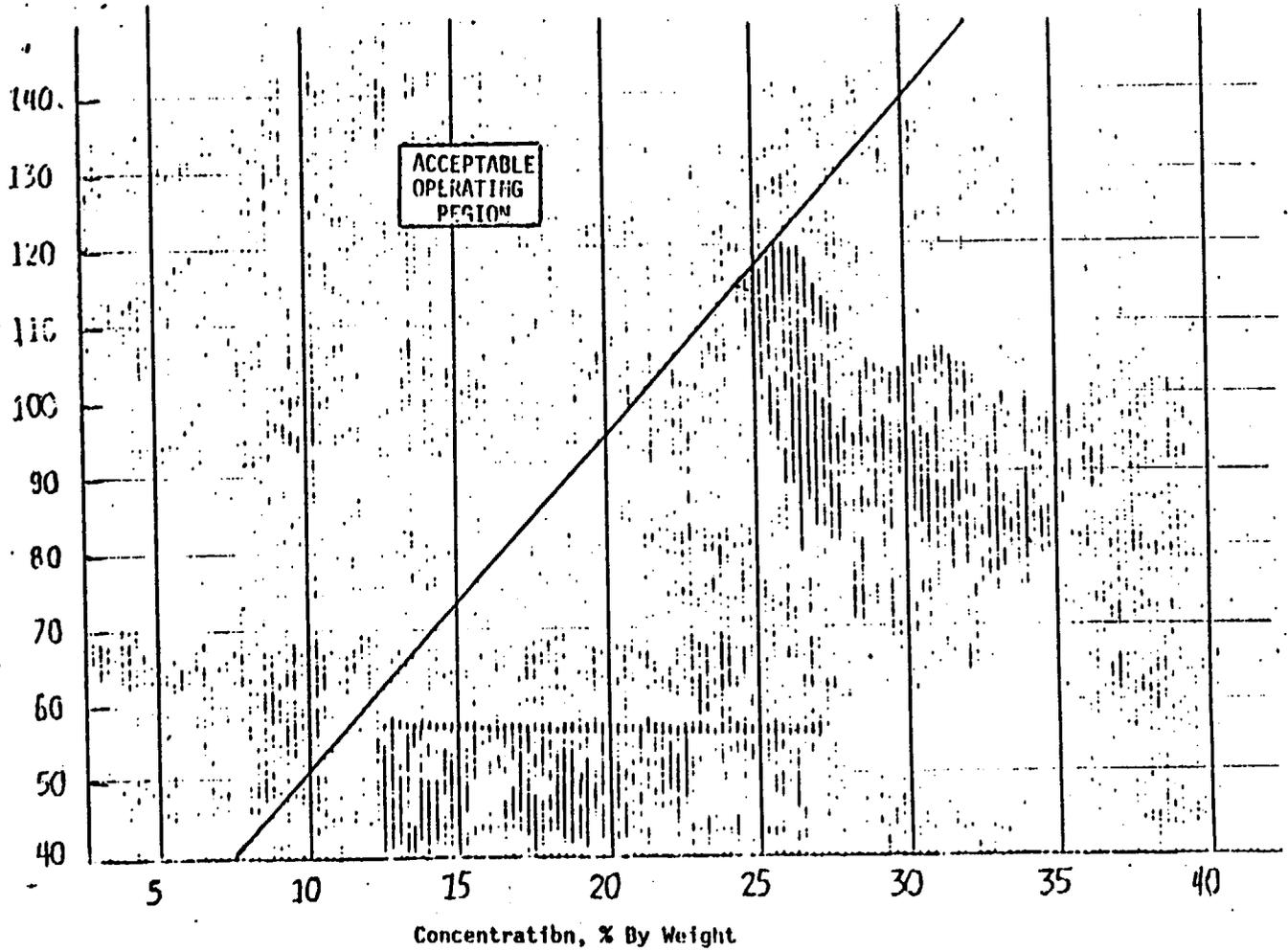
SR 3.1.7.5

\*This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below the limit of Figure 3.1.5-1.

\*\*This test shall also be performed whenever the heat tracing circuit has been found to be inoperable and may be performed by any series of sequential, overlapping or total flow path steps such that the entire flow path is included. A.5 A.6

LA SALLE - UNIT 1

3/4 1-21



SODIUM PENTABORATE SOLUTION TEMPERATURE/CONCENTRATION REQUIREMENTS

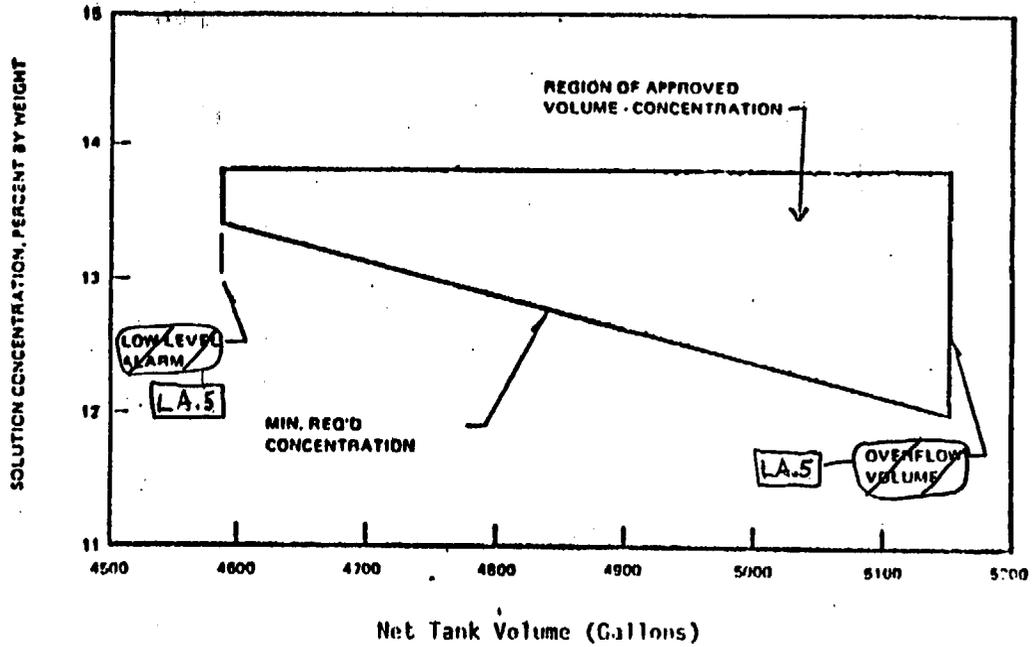
Figure 3.1.5-1

Figure 3.1.7-2

Page 3 of 8

A.1

ITS 3.1.7



Sodium Pentaborate ( $\text{Na}_2\text{B}_{10}\text{O}_{16} \cdot 10 \text{H}_2\text{O}$ )  
Volume/Concentration Requirements

Figure 3.1.5-2

Figure 3.1.7-1

A.1

REACTIVITY CONTROL SYSTEM

A.1

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.1.7

3.1.5 The standby liquid control system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 5<sup>a</sup>

L.1

ACTION:

a. In OPERATIONAL CONDITION 1 or 2:

ACTION A

1. With one motor operated suction valve, one pump and/or one explosive valve inoperable, restore the inoperable suction valve, pump and/or explosive valve to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION C

ACTION B

ACTION C

2. With the standby liquid control system inoperable, restore the system to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5<sup>a</sup>:

1. With one motor operated suction valve, one pump and/or one explosive valve inoperable, restore the inoperable suction valve, pump and/or explosive valve to OPERABLE status within 30 days or insert all insertable control rods within the next hour.

L.1

2. With the standby liquid control system inoperable, insert all insertable control rods within 1 hour.

SURVEILLANCE REQUIREMENTS

4.1.5 The standby liquid control system shall be demonstrated OPERABLE:

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

SR 3.1.7.1

SR 3.1.7.2

SA 3.1.7.3

1. The available volume and temperature of the sodium pentaborate solution are within the limits of Figures 3.1.5-1 and 3.1.5-2, and

2. The heat tracing circuit is OPERABLE by verifying the indicated temperature to be  $\geq 80^{\circ}\text{F}$  (on the local indicator)

A.2

of the pump suction piping up to the storage tank outlet valves are

With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

L.1

REACTIVITY CONTROL SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days by;

- 1. Starting both pumps and recirculating demineralized water to the test tank. L.2
- SR 3.1.7.4 2. Verifying the continuity of the explosive charge.
- SR 3.1.7.5 3. Determining that the concentration of boron in solution is within the limits of Figure 3.1.5-2 by chemical analysis.\*
- SR 3.1.7.6 4. Verifying that each valve in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

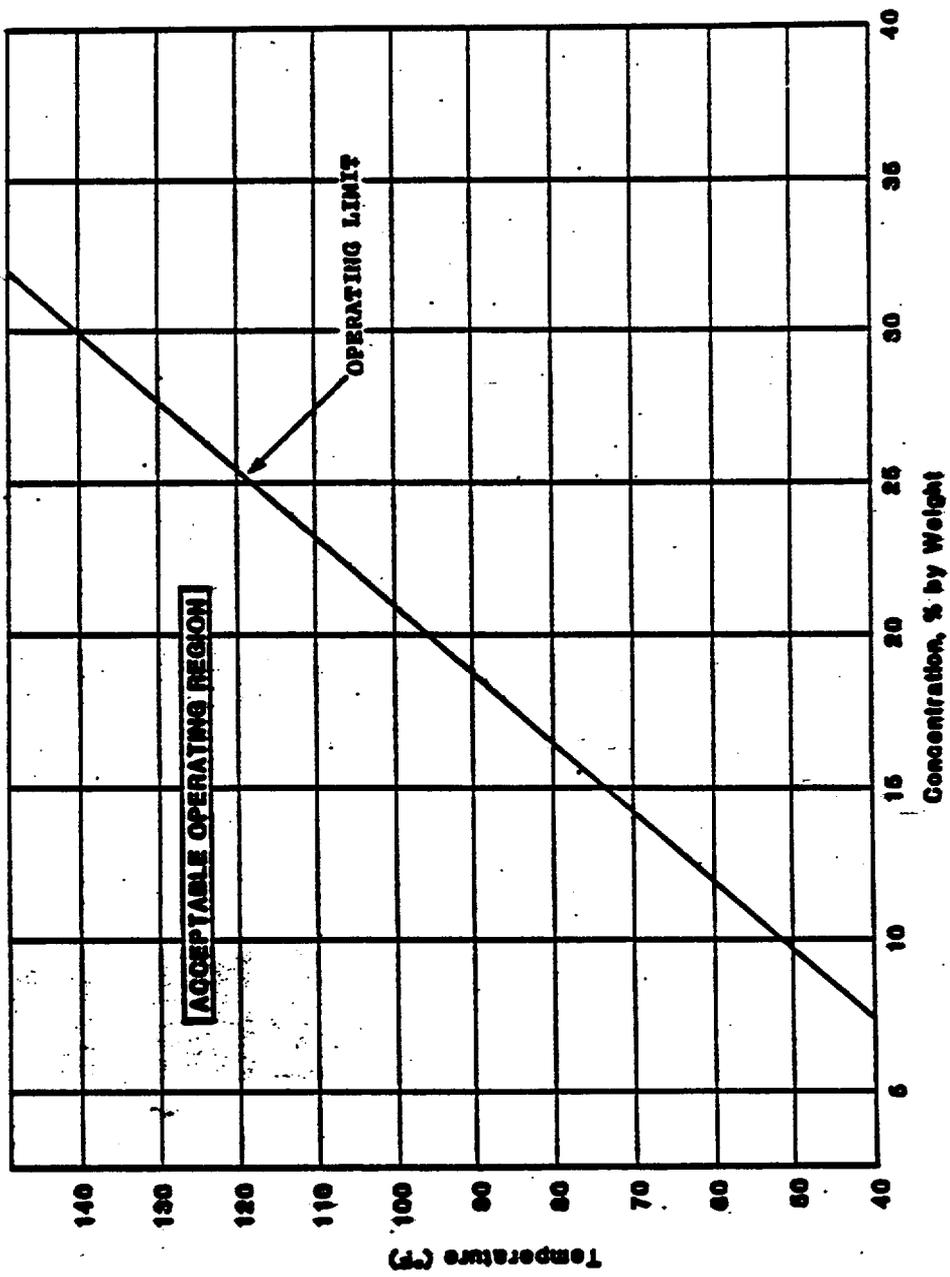
c. At least once per 18 months (during shutdown) by;

- 1. Initiating one of the standby liquid control system loops, including an explosive valve and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of that batch successfully fired. Both injection loops shall be tested in 48 months. L.3, A.3, L.A.2, L.D.1
- SR 3.1.7.8 L.A.2
- Definition of STRAGEEADTEST BASIS
- SR 3.1.7.7 2. Demonstrating that when tested pursuant to Specification 4.0.5, the minimum flow requirement of 41.2 gpm at a pressure of greater than or equal to 1220 psig is met.
- 3. Demonstrating that the pump relief valve setpoint is less than or equal to 1400 psig and verifying that the relief valve does not actuate during recirculation to the test tank. L.4, A.4
- A.4
- Storage tank outlet valves are
- SR 3.1.7.9 4. Demonstrating that all heat traced piping between the storage tank and the reactor vessel is unblocked by verifying flow from the storage tank to the motor operated suction valve and then draining and flushing the piping with demineralized water. A.5, L.A.3
- 5. Demonstrating that the storage tank heaters are OPERABLE by verifying the expected temperature rise for the sodium pentaborate solution in the storage tank after the heaters are energized. L.4, A.8, Storage tank outlet

SR 3.1.7.5 \*This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below the limit of Figure 3.1.5-1.

\*\*This test shall also be performed whenever the heat tracing circuit has been found to be inoperable and may be performed by any series of sequential, overlapping or total flow path steps such that the entire flow path is included. A.5

A.1

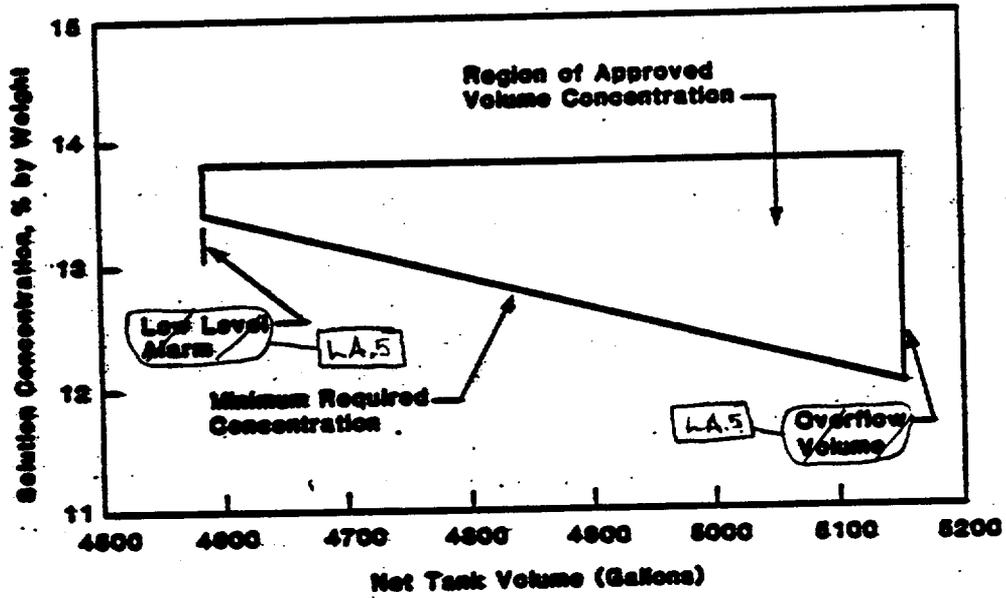


SODIUM PENTABORATE SOLUTION TEMPERATURE / CONCENTRATION REQUIREMENTS

Figure 3.1.5-1

Figure 3.1.7-2

A.1



**SODIUM PENTABORATE (Na<sub>2</sub>B<sub>10</sub>O<sub>16</sub> · 10H<sub>2</sub>O)  
VOLUME/CONCENTRATION REQUIREMENTS**

Figure 3.1.5-2

Figure 3.1.7-1

DISCUSSION OF CHANGES  
ITS: 3.1.7 - STANDBY LIQUID CONTROL SYSTEM

ADMINISTRATIVE

- A.1 In the conversion of the LaSalle 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-1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).
- A.2 CTS 4.1.5.a.2 requires the heat tracing to be determined OPERABLE by verifying the temperature to be  $\geq 60^{\circ}\text{F}$ . This has been clarified in proposed SR to require the temperature of "the suction piping up to the tank outlet valve" to be  $\geq 68^{\circ}\text{F}$  (see Discussion of Change M.1 for the change to the lower temperature limit), consistent with the current manner in which CTS 4.1.5.a.2 is performed. The storage tank outlet valve is a normally closed valve and the piping is only heat traced up to this valve. Also, since heat tracing is the system that provides heat to the piping, it is not necessary to mention in the SR. In addition, due to this design, CTS 4.1.5.c.4 requires the piping to be flushed after the SLC pump is used to pump boron from the storage tank to preclude boron precipitation in the non-heat traced piping. Therefore, this addition is considered administrative.
- A.3 CTS 4.1.5.b.4 requires a verification that each valve in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position. The SLC system is required to be capable of being manually activated by 10 CFR 50.62. In current Technical Specifications, it is recognized and interpreted that "in the correct position" allows the valves to be in a non-accident position provided they can be realigned to the correct position. In the ITS, the words "in the correct position" mean that the valves must be in the accident position, unless they can be automatically aligned on an accident signal. If so, then they can be in the non-accident position. Thus, for the SLC System the additional words "or can be aligned to the correct position" have been added in SR 3.1.7.6 to clarify that it is permissible for this systems' valves to be in the non-accident position and still be considered OPERABLE. Since this is the current requirement, this change is considered administrative.
- A.4 CTS 4.1.5.c.4 (proposed SR 3.1.7.9) verifies that all heat traced piping between the storage tank and the reactor vessel is unblocked. Since heat traced piping exists only between the storage tank and the pump suction (storage tank outlet) valve in the SLC System, and the second part of the sentence requires pumping only to the test tank, the Surveillance has been reworded, for clarity, to identify the extent of the system heat traced piping. This is consistent with the current requirement, and as such, is considered an administrative change only.

DISCUSSION OF CHANGES  
ITS: 3.1.7 - STANDBY LIQUID CONTROL SYSTEM

ADMINISTRATIVE (continued)

- A.5 CTS 4.1.5.c.4 footnote \*\* requires the heat traced piping to be demonstrated unblocked whenever the heat tracing circuit has been found to be inoperable. The Frequency of CTS 4.1.5.c.4 footnote \*\* has been changed such that the Surveillance is required if the piping temperature drops below the lower limit, similar to footnote \* to CTS 4.1.5.b.3. Since the intent is to ensure no piping is blocked, and the temperature is the best indicator of a condition in which the piping can become blocked, this change is considered administrative in nature.
- A.6 CTS 4.1.5.c.4 requires a demonstration that the heat traced piping is unblocked by pumping from the storage tank to the test tank. Footnote \*\* to this Surveillance states that it can be performed by any series of sequential, overlapping or total flow path steps such that the entire flow path is included. The allowance is unnecessary since the test can only be performed in one step; by pumping from the storage tank to the test tank. Therefore, this allowance has been deleted and its deletion is considered administrative.
- A.7 CTS Surveillance Requirement 4.1.5.c.4 requires demonstration that "all heat traced piping between the storage tank and the reactor vessel is unblocked by verifying flow from the storage tank to the motor operated suction valve and then draining and flushing the piping with demineralized water." The Standby Liquid Control system piping is heat traced from the point where it exits the storage tank up to the motor operated valve that is located on the suction line for each standby liquid control pump. Station operating procedures refer to this motor operated valve as the storage tank outlet valve. Downstream of each storage tank outlet valve there is also a locked-open manual valve that is procedurally referred to as the pump suction valve. The piping between the storage tank outlet valve and the pump suction valve does not contain sodium pentaborate solution when the system is in its normal, standby condition, and is therefore not heat traced. Consistent with the existing plant nomenclature, references in CTS 4.1.5.4.c.4 to the "motor operated suction valve" have been changed throughout ITS 3.1.7 to reflect the "storage tank outlet valve."

DISCUSSION OF CHANGES  
ITS: 3.1.7 - STANDBY LIQUID CONTROL SYSTEM

TECHNICAL CHANGES - MORE RESTRICTIVE

- M.1 CTS 4.1.5.a.2 requires verification that the temperature of the pump suction piping (up to the pump suction (storage tank outlet) valve) is  $\geq 60^{\circ}\text{F}$ . This temperature is below that specified in CTS Figure 3.1.5-2 at the maximum allowable sodium pentaborate solution concentration. Therefore, the pump suction piping temperature limit is increased to  $\geq 68^{\circ}\text{F}$ , which is consistent with the temperature/concentration requirements of CTS Figure 3.1.5-1 at the maximum allowable sodium pentaborate solution concentration. This change represents an additional restriction on plant operation necessary to ensure boron remains in solution.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA.1 The CTS 4.1.5.a.2 detail for performance of the surveillance (verifying temperature "on the local indicator") 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 3.1.7.3 are adequate to ensure the temperature of the suction piping is maintained to support flow to each SLC subsystem 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 details of the method for performing CTS 4.1.5.c.1, the Surveillance to verify flow through the SLC subsystem into the reactor pressure vessel (initiating an explosive valve and the requirements on the replacement charges for explosive valve), are proposed to be relocated to the Bases. These details are not necessary to ensure that 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 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.3 The details of the method for performing CTS 4.1.5.c.1, the Surveillance to verify flow through the SLC subsystem into the reactor pressure vessel (by pumping demineralized water into the reactor vessel), are proposed to be relocated to the Bases. These details are not necessary to ensure that SLC System is maintained OPERABLE. The requirements of ITS 3.1.7 and

DISCUSSION OF CHANGES  
ITS: 3.1.7 - STANDBY LIQUID CONTROL SYSTEM

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.3 (cont'd) 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 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.

The details of the method for performing CTS 4.1.5.c.4, the Surveillance to demonstrate all piping between the SLC storage tank and the pump suction is unblocked (by pumping from the storage tank to the test tank and then draining and flushing the piping with demineralized water), are proposed to be relocated to the Bases. These details are not necessary to ensure that SLC System is maintained OPERABLE. The requirements of ITS 3.1.7 and SR 3.1.7.9 are adequate to ensure the piping between the SLC storage tank and the pump suction is unblocked and to ensure SLC System OPERABILITY. 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.4 The testing requirements of CTS 4.1.5.c.3 for SLC System relief valve setting verification is proposed to be relocated to the Inservice Testing (IST) Program. These testing requirements do demonstrate the SLC System relief valves are OPERABLE. However, the IST Program, required by 10 CFR 50.55a, provides requirements for the testing of all ASME Code Class 1, 2, and 3 valves in accordance with applicable codes, standards, and relief requests, endorsed by the NRC for LaSalle 1 and 2. Compliance with 10 CFR 50.55a, and as a result the IST Program and implementing procedures, is required by the LaSalle 1 and 2 Operating Licenses. These controls are adequate to ensure the required testing to demonstrate OPERABILITY is performed. Therefore, the relocated requirements are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the relocated requirements in the IST Program will be controlled by the provisions of 10 CFR 50.59 and 10 CFR 50.55a.

LA.5 The contained tank volumes associated with the SLC storage tank low level and high level alarms are details of the system design and are proposed to be relocated to the UFSAR. These design details are not necessary to ensure the OPERABILITY of the SLC System. SLC System OPERABILITY requirements are adequately addressed in ITS 3.1.7 and the definition of OPERABILITY. 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 UFSAR will be controlled by the provisions of 10 CFR 50.59.

DISCUSSION OF CHANGES  
ITS: 3.1.7 - STANDBY LIQUID CONTROL SYSTEM

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

- LD.1 The Frequency for performing CTS 4.1.5.c.1 and 4.1.5.c.4 (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 4.1.5.c.1) (i.e., a maximum of 22.5 months (45 months for CTS 4.1.5.c.1) accounting for the allowable grace period specified in CTS 4.0.2 and proposed SR 3.0.2) to a 24 month Surveillance Frequency (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 4.0.2 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 small. 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 small. 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 small from a change to CTS 4.1.5.c.1 and 4.1.5.c.4 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.

DISCUSSION OF CHANGES  
ITS: 3.1.7 - STANDBY LIQUID CONTROL SYSTEM

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

"Specific"

- L.1 CTS 3.1.5 Applicability requires the SLC System to be OPERABLE in MODE 5 with any control rod withdrawn. In ITS 3.1.7 the SLC System is proposed not to be required during refueling since only a single control rod can be withdrawn (ITS 3.9.2 enforces the one-rod-out interlock) and adequate SHUTDOWN MARGIN (enforced by ITS 3.1.1) prevents criticality under these conditions.
- L.2 CTS 4.1.5.b.1, which requires testing both SLC pumps every 31 days, is proposed to be deleted. Testing the SLC pumps in accordance with the Inservice Testing Program, 10 CFR 50.55a, per CTS 4.1.5.c.2 (proposed SR 3.1.7.7) provides adequate assurance that the pumps will perform their required function. More frequent testing is not warranted based on surveillance history. Deletion of this requirement is a less restrictive change.
- L.3 The requirement that the testing in CTS 4.1.5.c (proposed SR 3.1.7.7, SR 3.1.7.8, and SR 3.1.7.9) be performed "during shutdown" is proposed to be deleted. This Surveillance might be able to be performed while operating without jeopardizing safe plant operation. The control of plant conditions appropriate to perform the test is an issue for procedures and scheduling and has been determined by the NRC staff to be unnecessary as a Technical Specification restriction. As indicated in Generic Letter 91-04, allowing this control is consistent with the vast majority of other Technical Specification Surveillances that do not dictate plant conditions for the Surveillance. This detail of the Surveillance is a prerequisite for performance of the test and is not necessary for ensuring the requirements of SR 3.1.7.7, SR 3.1.7.8, and SR 3.1.7.9 adequately demonstrate the OPERABILITY of the SLC System.
- L.4 CTS 4.1.5.c.5, which requires the heaters to be demonstrated OPERABLE by verifying a temperature rise in the storage tank, has been deleted. The daily verification of the solution temperature, CTS 4.1.5.a.1 (proposed SR 3.1.7.2), provides an adequate check on the capability of the storage tank heaters to maintain solution temperature, when necessary. The heaters are designed to automatically maintain the solution temperature > 75°F. With temperature > 75°F, the boron in solution cannot precipitate out. If the ambient temperature in the reactor building is sufficient to maintain the temperature > 75°F, then the heaters are not needed to prevent boron precipitation. When ambient temperature in the reactor building is not sufficient to maintain solution temperature > 75°F, the heaters are needed and, if not functioning properly, the daily temperature verification will ensure that this does not go unnoticed. CTS 4.1.5.c.5 is only required once per 18 months, and as such, the daily verification of solution temperature currently provides adequate indications of heater

DISCUSSION OF CHANGES  
ITS: 3.1.7 - STANDBY LIQUID CONTROL SYSTEM

TECHNICAL CHANGES - LESS RESTRICTIVE

L.4 Operability (the ambient temperature in the reactor building is not continuously  
(cont'd) high enough to maintain solution temperature  $> 75^{\circ}\text{F}$  without heaters over an 18  
month cycle).

RELOCATED SPECIFICATIONS

None

REACTIVITY CONTROL SYSTEM

add proposed LCO and Applicability

A.2

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

See ITS 3.1.3

- 2. If the inoperable control rod(s) is inserted:
  - a) Within 1 hour disarm the associated directional control valves either:
    - 1) Electrically, or
    - 2) Hydraulically by closing the drive water and exhaust water isolation valves.
  - b) Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.
- c. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

- ACTION A { d\*. With one or more SDV vent or drain lines with one valve inoperable,
  - 1. Isolate\*\* the associated line within 7 days.
  - 2. Otherwise, be in HOT SHUTDOWN within the next 12 hours.

- ACTION B { e\*. With one or more SDV vent or drain lines with both valves inoperable,
  - 1. Isolate\*\* the associated line within 8 hours.
  - 2. Otherwise, be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

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

- SR3.1.8.1 a. At least once per 31 days verifying each valve to be open\*\*, and
- SR3.1.8.2 b. At least once per 92 days cycling each valve through at least one complete cycle of full travel.

4.1.3.1.2 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:

See ITS 3.1.3

- a. At least once per 7 days, and
- b. At least once per 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference.

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

Note to SR3.1.8.1 - \*\*These valves may be closed intermittently for testing under administrative control.

Note 1 to Actions - \*\*Separate Action statement entry is allowed for each SDV vent and drain line. An isolated line may be unisolated under administrative control to allow draining and venting of the SDV.

Note 2 to Actions

REACTIVITY CONTROL SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

4.1.3.1.3 All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.1.3.2, 4.1.3.4, 4.1.3.5, 4.1.3.6 and 4.1.3.7.

(See ITS 3.1.3)

SR3.18.3

4.1.3.1.4 The scram discharge volume shall be determined OPERABLE by demonstrating the scram discharge volume drain and vent valves OPERABLE at least once per 12 months by verifying that the drain and vent valves:

24

- a. Close within 30 seconds after receipt of a signal for control rods to scram, and
- b. Open after the scram signal is reset.

LD.1

actual or simulated

A-3

A.1

REACTIVITY CONTROL SYSTEM

add proposed LCO and Applicability A.2

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

See ITS 3.1.3

- 2. If the inoperable control rod(s) is inserted:
  - a) Within 1 hour disarm the associated directional control valves<sup>1</sup> either:
    - 1) Electrically, or
    - 2) Hydraulically by closing the drive water and exhaust water isolation valves.
  - b) Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.
  - c. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

- ACTION A { d<sup>1</sup>. With one or more SDV vent or drain lines with one valve inoperable,
  - 1. Isolate<sup>2</sup> the associated line within 7 days.
  - 2. Otherwise, be in HOT SHUTDOWN within the next 12 hours.

- ACTION B { e<sup>1</sup>. With one or more SDV vent or drain lines with both valves inoperable,
  - 1. Isolate<sup>2</sup> the associated line within 8 hours.
  - 2. Otherwise, be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

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

- SR 3.1.8.1 a. At least once per 31 days verifying each valve to be open<sup>3</sup>, and
- SR 3.1.8.2 b. At least once per 92 days cycling each valve through at least one complete cycle of full travel.

4.1.3.1.2 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. At least once per 7 days, and
- b. At least once per 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference.

See ITS 3.1.3

<sup>1</sup>May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

Note to SR 3.1.8.1

<sup>2</sup>These valves may be closed intermittently for testing under administrative control.

Note 1 to Actions

<sup>3</sup>Separate Action statement entry is allowed for each SDV vent and drain line.

<sup>4</sup>An isolated line may be unisolated under administrative control to allow draining and venting of the SDV.

Note 2 to Actions

A.1

ITS 3.1.8

REACTIVITY CONTROL SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

4.1.3.1.3 All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.1.3.2, 4.1.3.4, 4.1.3.5, 4.1.3.6 and 4.1.3.7.

See ITS 3.1.3

SR3.1.8.3 4.1.3.1.4 The scram discharge volume shall be determined OPERABLE by demonstrating the scram discharge volume drain and vent valves OPERABLE at least once per 18 months by verifying that the drain and vent valves:

- a. Close within 30 seconds after receipt of a signal for control rods to scram, and
- b. Open after the scram signal is reset.

LD-1

actuator simulated

A.3

DISCUSSION OF CHANGES  
ITS: 3.1.8 - SDV VENT AND DRAIN VALVES

ADMINISTRATIVE

- A.1 In the conversion of the LaSalle 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-1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).
- A.2 The scram discharge volume (SDV) vent and drain valve requirements in CTS 3/4.1.3.1 have been moved to a separate Specification. As such, an LCO and Applicability statement have been added (ITS 3.1.8). The ITS 3.1.8 LCO and Applicability requirements are consistent with the current requirements of CTS 3/4.1.3.1, therefore this addition is considered administrative.
- A.3 The phrase "actual or simulated" in reference to the signal used for performing CTS 4.1.3.1.4.a and CTS 4.1.3.1.4.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 4.1.3.1.4 (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  $\leq 30$  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 4.0.2 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 4.0.2 and proposed

DISCUSSION OF CHANGES  
ITS: 3.1.8 - SDV VENT AND DRAIN VALVES

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd) 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 small. 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.

DISCUSSION OF CHANGES  
ITS: 3.1.8 - SDV VENT AND DRAIN VALVES

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

"Specific"

None

RELOCATED SPECIFICATIONS

None

REACTIVITY CONTROL SYSTEM

CONTROL ROD DRIVE HOUSING SUPPORT

LIMITING CONDITION FOR OPERATION

3.1.3.8 The control rod drive housing support shall be in place.

APPLICABILITY: OPERATIONAL CONDITIONS 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 COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.8 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.

L1

REACTIVITY CONTROL SYSTEM

CONTROL ROD DRIVE HOUSING SUPPORT

LIMITING CONDITION FOR OPERATION

3.1.3.8 The control rod drive housing support shall be in place.

APPLICABILITY: OPERATIONAL CONDITIONS 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 COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.8 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: 3/4.1.3.8 - CONTROL ROD DRIVE HOUSING SUPPORT

ADMINISTRATIVE

None

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

None

"Specific"

- L.1        The CTS 3/4.1.3.8 requirement for the CRD 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

REACTIVITY CONTROL SYSTEM

3/4.1.6 ECONOMIC GENERATION CONTROL SYSTEM

R.1

LIMITING CONDITION FOR OPERATION

3.1.6 The economic generation control system may be in operation with automatic flow control provided that:

- a. Core flow is  $\geq$  65% of rated core flow, and
- b. THERMAL POWER is greater than or equal to 20% of RATED THERMAL POWER.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION: With core flow less than 65% of rated core flow or THERMAL POWER less than 20% of RATED THERMAL POWER, cease operation under the economic generation control system.

SURVEILLANCE REQUIREMENTS

4.1.6 The economic generation control system shall be demonstrated OPERABLE by:

- a. Calculating current efficiency and, using a nominal curve of efficiency versus THERMAL POWER, verifying that the EGC lower MW setpoint will maintain core flow  $>$  65% of rated core flow and THERMAL POWER  $\geq$  20% of RATED THERMAL POWER:
  - 1. Prior to entry into EGC operation, and
  - 2. At least once per 12 hours while operating in EGC.
- b. Verifying that current core flow is  $>$  65% of rated core flow and THERMAL POWER is  $\geq$  20% of RATED THERMAL POWER:
  - 1. Prior to entry into EGC operation, and
  - 2. At least once per 12 hours while operating in EGC.

REACTIVITY CONTROL SYSTEM

3/4.1.6 ECONOMIC GENERATION CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.6 The economic generation control system may be in operation with automatic flow control provided that:

- a. Core flow is  $\geq$  65% of rated core flow, and
- b. THERMAL POWER is greater than or equal to 20% of RATED THERMAL POWER.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION: With core flow less than 65% of rated core flow or THERMAL POWER less than 20% of RATED THERMAL POWER, cease operation under the economic generation control system.

SURVEILLANCE REQUIREMENTS

4.1.6 The economic generation control system shall be demonstrated OPERABLE by:

- a. Calculating current efficiency and, using a nominal curve of efficiency versus THERMAL POWER, verifying that the EGC lower MW setpoint will maintain core flow  $>$  65% of rated core flow and THERMAL POWER  $\geq$  20% of RATED THERMAL POWER:
  - 1. Prior to entry into EGC operation, and
  - 2. At least once per 12 hours while operating in EGC.
- b. Verifying that current core flow is  $>$  65% of rated core flow and THERMAL POWER is  $\geq$  20% of RATED THERMAL POWER:
  - 1. Prior to entry into EGC operation, and
  - 2. At least once per 12 hours while operating in EGC.

R.1

DISCUSSION OF CHANGES  
CTS: 3/4.1.6 - ECONOMIC GENERATION CONTROL SYSTEM

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 3/4.1.6 did not satisfy the NRC Final Policy Statement Technical Specification screening criteria as documented in the Application Selection Criteria to the LaSalle 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 1-1 through B 3/4 1-6) have been completely replaced by revised Bases reflecting the format and applicable content of the LaSalle 1 and 2 ITS Section 3.1, consistent with the BWR ISTS, NUREG-1434, Rev. 1. The revised Bases are as shown in the LaSalle 1 and 2 ITS Bases.

<CTS>

3.1 REACTIVITY CONTROL SYSTEMS  
3.1.1 SHUTDOWN MARGIN (SDM)

<LCO 3.1.1>

LCO 3.1.1

SDM shall be:

- a.  $\geq 70.38\% \Delta k/k$ , with the highest worth control rod analytically determined; or
- b.  $\geq 40.28\% \Delta k/k$ , with the highest worth control rod determined by test.

1

<Appl 3.1.1>

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

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.1.1 Act a>	A. SDM not within limits in MODE 1 or 2.	A.1 Restore SDM to within limits.	6 hours
<3.1.1 Act a>	B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	12 hours
<3.1.1 Act b>	C. SDM not within limits in MODE 3.	C.1 Initiate action to fully insert all insertable control rods.	Immediately
<3.1.1 Act b>	D. SDM not within limits in MODE 4.	D.1 Initiate action to fully insert all insertable control rods.	Immediately
		<u>AND</u>	(continued)

<CTS>

ACTIONS		
CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>&lt;3.1.1 Actb&gt; D. (continued)</p> <p style="text-align: center;">2</p>	<p>D.2 Initiate action to restore <del>secondary containment</del> to OPERABLE status.</p>	1 hour
	<p>AND</p> <p>D.3 <del>*</del> Initiate action to restore one standby gas treatment (SGT) subsystem to OPERABLE status.</p>	1 hour
	<p>AND</p> <p>D.4 Initiate action to restore isolation capability in each required <del>secondary containment</del> penetration flow path not isolated.</p>	1 hour
<p>&lt;3.1.1 Actc&gt; E. SDM not within limits in MODE 5.</p>	<p>E.1 Suspend CORE ALTERATIONS except for control rod insertion and fuel assembly removal.</p> <p>AND</p>	<p>Immediately</p> <p>(continued)</p>

<CTS>

ACTIONS		
CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>&lt;3.1.1 Actc&gt; E. (continued)</p> <p style="text-align: center;">2</p>	<p>E.2 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.</p>	Immediately
	<p>AND</p>	
	<p>E.3 Initiate action to restore <del>secondary containment</del> to OPERABLE status.</p>	1 hour
	<p>AND</p>	
	<p>E.4 Initiate action to restore one SGT subsystem to OPERABLE status.</p>	1 hour
<p>AND</p>		
	<p>E.5 Initiate action to restore isolation capability in each required <del>secondary containment</del> penetration flow path not isolated.</p>	1 hour

<CTS>

**SURVEILLANCE REQUIREMENTS**

<4.1.1. a  
DOC M.1>

SURVEILLANCE	FREQUENCY
<p>SR 3.1.1.1 Verify SDM is:</p> <div style="border: 1px solid black; padding: 5px; margin-left: 20px;"> <p>a. <math>\geq 0.38\% \Delta k/k</math> with the highest worth control rod analytically determined; or</p> <p>b. <math>\geq 0.28\% \Delta k/k</math> with the highest worth control rod determined by test.</p> </div>	<p>Prior to each in vessel fuel movement during fuel loading sequence</p> <p><b>AND</b></p> <p>Once within 4 hours after criticality following fuel movement within the reactor pressure vessel or control rod replacement</p>

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1  
ITS: 3.1.1 - SHUTDOWN MARGIN

1. TSTF-9 relocates SHUTDOWN MARGIN (SDM) limits of NUREG-1434 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 LaSalle 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

<LCO 3.1.2>

LCO 3.1.2

The reactivity ~~difference~~ between the ~~monitored~~ core  $k_{eff}$  and the predicted core  $k_{eff}$  shall be within  $\pm 1\% \Delta k/k$ .

<Appl 3.1.2>

APPLICABILITY: MODES 1 and 2.

ACTIONS

<3.1.2 Act a>

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

<3.1.2 Act b>

<CTS>

**SURVEILLANCE REQUIREMENTS**

<4.1.2>

SURVEILLANCE	FREQUENCY
<p>SR 3.1.2.1</p> <p>1 } Verify core reactivity {difference} between the monitored core <math>k_{eff}</math> and the predicted core <math>k_{eff}</math> is within <math>\pm 1\% \Delta k/k</math>.</p>	<p>Once within 24 hours after reaching equilibrium conditions following startup after fuel movement within the reactor pressure vessel or control rod replacement</p> <p><b>AND</b></p> <p>1000 MWD/T thereafter during operations in MODE 1</p>

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1  
ITS: 3.1.2 - REACTIVITY ANOMALIES

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

<LCO 3.1.3.1> LCO 3.1.3 Each control rod shall be OPERABLE.

<Appl 3.1.3.1> APPLICABILITY: MODES 1 and 2.

ACTIONS

<DOC A.3> -----NOTE-----  
Separate Condition entry is allowed for each control rod.  
-----

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>&lt;DOC A.4&gt; &lt;DOC L.3&gt; A. One withdrawn control rod stuck.</p>	<p>Rod Worth Minimizer (RWM)</p> <p>-----NOTE----- A stuck rod may be bypassed in the Rod Action Control System (RACS) in accordance with SR 3.3.2.1.8 if required to allow continued operation.</p> <p>A.1 2 AND</p> <p>TSTF-32</p>	<p>1</p> <p>as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation,"</p> <p>2 hours</p> <p>(continued)</p>

<3.1.3.1 Act a.  
3.1.3.1 Act a.1.b)  
3.1.3.1 Act a.1.c)>  
<4.1.3.1.2.6>  
<4.1.1  
4.1.1.c)>  
<3.1.3.2 Act>

A.1 Verify stuck control rod separation criteria are met. Immediately

AND

<CTS>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. (continued)</p> <p>&lt;DOC A.4&gt; &lt;DOC L.3&gt;</p> <p>&lt;3.1.3.1 Act a 3.1.3.1 Act a.1.b 3.1.3.1 Act a.1.c&gt;</p> <p>&lt;4.1.3.1.2.b&gt;</p> <p>&lt;4.1.1&gt;</p> <p>&lt;4.1.1.c&gt;</p> <p>&lt;3.1.3.2 Act&gt;</p>	<p>A. ①</p> <p>TSTF-32 ③</p> <p><del>-----NOTE----- Not applicable when less than or equal to the low power setpoint (LPSP) of the Rod Pattern Control System (RPCS).</del></p> <p>Perform SR 3.1.3.2 and SR 3.1.3.3 for each withdrawn OPERABLE control rod.</p> <p>AND ④ TSTF-32</p> <p>A. ② ① Perform SR 3.1.1.1.</p>	<p>TSTF-33</p> <p>24 hours</p> <p>72 hours</p> <p>from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM</p> <p>1</p>
<p>&lt;DOC M.2&gt; B. Two or more withdrawn control rods stuck.</p>	<p>B.1 Disarm the associated CRD.</p> <p>AND</p> <p>B. ② ① Be in MODE 3.</p>	<p>2 hours</p> <p>12 hours</p> <p>TSTF-34</p>

(continued)

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>&lt;3.1.3.1 Act b&gt;c. One or more control rods inoperable for reasons other than Condition A or B.</p> <p>3.1.3.1 Act b.1.a)2) 3.1.3.1 Act b.1.b 3.1.3.1 Act b.2 3.1.3.1 Act b.2.a)</p> <p>&lt;3.1.3.2 Act&gt; &lt;3.1.3.6 Act a.1&gt; &lt;3.1.3.7 Act a&gt;</p>	<p>C.1</p> <p>-----NOTE----- Inoperable control rods may be bypassed in RACS in accordance with SR 3.3.2.1.B, if required, to allow insertion of inoperable control rod and continued operation.</p> <p>-----</p> <p>Fully insert inoperable control rod.</p> <p>AND</p> <p>C.2 Disarm the associated CRD.</p>	<p>RWM 1</p> <p>as allowed by LCO 3.3.2.1</p> <p>3 hours</p> <p>4 hours</p>
<p>D. -----NOTE----- Not applicable when THERMAL POWER &gt; 10% RTP.</p> <p>2</p> <p>&lt;3.1.3.1 Act a.1.a&gt; &lt;3.1.3.1 Act b.1.a&gt;</p> <p>Two or more inoperable control rods not in compliance with banked position withdrawal sequence (BPWS) and not separated by two or more OPERABLE control rods.</p>	<p>D.1 Restore compliance with BPWS</p> <p>OR</p> <p>D.2 Restore control rod to OPERABLE status.</p> <p>analyzed rod position sequence</p>	<p>4 hours</p> <p>4 hours</p> <p>3</p>

(continued)

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>E. -----NOTE----- Not applicable when THERMAL POWER &gt; [10]% RTP. -----</p> <p>One or more groups with four or more inoperable control rods.</p>	<p>E.1 Restore the control rod to OPERABLE status.</p>	<p>4 hours</p>
<p>3.1.3.1 Act a.2 3.1.3.1 Act b.2.b 3.1.3.1 Act c</p> <p>Required Action and associated Completion Time of Condition A, C, D, or E not met.</p> <p>OR</p> <p>3.1.3.6 Act a.2 3.1.3.7 Act a.4</p> <p>Nine or more control rods inoperable.</p>	<p>F.1 Be in MODE 3.</p>	<p>12 hours</p>

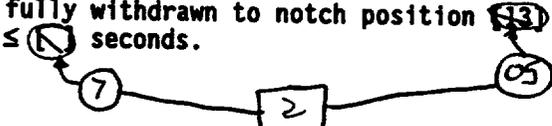
SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>&lt;LCO 3.1.3.7&gt; SR 3.1.3.1 Determine the position of each control rod. &lt;4.1.3.7&gt;</p>	<p>24 hours</p>

(continued)

<CTS>

**SURVEILLANCE REQUIREMENTS (continued)**

SURVEILLANCE	FREQUENCY
<p>&lt;4.1.3.1.2&gt; SR 3.1.3.2</p> <p>-----NOTE----- Not required to be performed until 7 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the <del>RPCS</del> <u>RWM</u> <u>1</u></p> <p>-----</p> <p>Insert each fully withdrawn control rod at least one notch.</p>	<p>7 days</p>
<p>&lt;4.1.3.1.2&gt; SR 3.1.3.3</p> <p>-----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 <del>RPCS</del> <u>RWM</u> <u>1</u></p> <p>-----</p> <p>Insert each partially withdrawn control rod at least one notch.</p>	<p>31 days</p>
<p>&lt;LCo 3.1.3.2&gt; &lt;DOC A.13&gt; SR 3.1.3.4</p> <p>Verify each control rod scram time from fully withdrawn to notch position <del>13</del> is <math>\leq</math> <u>7</u> seconds.</p> 	<p>In accordance with SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4</p>

(continued)

<CTS>

**SURVEILLANCE REQUIREMENTS (continued)**

<LCO 3.1.3.6>  
<4.1.3.6>

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

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1  
ITS: 3.1.3 - CONTROL ROD OPERABILITY

1. The LaSalle 1 and 2 rod pattern control design does not include a Rod Action Control System, but a rod worth minimizer (RWM), similar to the BWR/4 design. Therefore, the Notes have been modified to reflect the RWM design, and are consistent with the BWR/4 ISTS, NUREG-1433, Revision 1.
2. The brackets have been removed and the proper plant specific information/value has been provided.
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. ISTS 3.1.3 ACTION E is applicable to plants with Siemens Power Corporation (SPC) fuel. Although this station uses 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 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

DOC M.3 LCO 3.1.4

- a. No more than ~~(14)~~ <sup>(2)</sup> 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
3.1.3.3 Act 3.1.3.4 Act 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
4.1.3.2 4.1.3.2.a 4.1.3.3 4.1.3.4 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 $\geq$ <del>(950)</del> <sup>800</sup> psig.	<p>Prior to exceeding 40% RTP after fuel movement within the reactor pressure vessel.</p> <p>AND affected core cell</p> <p>(continued)</p>

TSTF-222

move to SR 3.1.4.4

<CTS>

**SURVEILLANCE REQUIREMENTS**

SURVEILLANCE	FREQUENCY
<p>SR 3.1.4.1 (continued)</p> <p>4.1.3.2 4.1.3.2.a 4.1.3.3 4.1.3.4</p>	<p>Prior to exceeding 40% RTP after each reactor shutdown <math>\geq</math> 120 days</p>
<p>SR 3.1.4.2</p> <p>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 <math>\geq</math> 800 psig.</p> <p>4.1.3.2 4.1.3.2.c 4.1.3.3 4.1.3.4</p>	<p>120 days cumulative operation in MODE 1</p>
<p>SR 3.1.4.3</p> <p>Verify each affected control rod scram time is within the limits of Table 3.1.4-1 with any reactor steam dome pressure.</p> <p>Doc M.2</p>	<p>Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect scram time</p>
<p>SR 3.1.4.4</p> <p>Verify each affected control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure <math>\geq</math> 800 psig.</p> <p>4.1.3.2 4.1.3.2.a 4.1.3.2.b 4.1.3.3 4.1.3.4</p>	<p><u>AND</u> (Insert Frequency from SR 3.1.4.1) Prior to exceeding 40% RTP after work on control rod or CRD System that could affect scram time</p> <p>TSTF-272</p>

Table 3.1.4-1  
Control Rod Scram Times

{DOC M.3}  
{LCO 3.1.3.3}  
{LCO 3.1.3.4}

- 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 > 2 seconds to notch position (23). These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

05  
1

NOTCH POSITION	SCRAM TIMES(a)(b) (seconds)	
	REACTOR STEAM DOME PRESSURE ≥ <u>(950)</u> psig	REACTOR STEAM DOME PRESSURE(c) [1058] psig
<u>(43)</u> <u>45</u>	<u>(0.30)</u> <u>0.41</u>	[0.31]
<u>(29)</u> <u>39</u>	<u>(0.78)</u> <u>0.80</u>	[0.84]
<u>(13)</u> <u>25</u>	<u>(1.40)</u> <u>1.77</u>	[1.53]
<u>05</u>	<u>3.20</u>	

- (a) Maximum scram time from fully withdrawn position based on de-energization of scram pilot valve solenoids as time zero.
- (b) Scram times as a function of reactor steam dome pressure when < 950 psig are within established limits.
- (c) For intermediate reactor steam dome pressures, the scram time criteria are determined by linear interpolation.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, 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. The Table has been modified since the LaSalle 1 and 2 safety analysis assumes only one set of scram times at one pressure, similar to the BWR/4 ISTS, NUREG-1433, Revision 1.
3. Editorial change for enhanced clarity.

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Control Rod Scram Accumulators

<LCO 3.1.3.5> LCO 3.1.5 Each control rod scram accumulator shall be OPERABLE.

<App 3.1.3.5> APPLICABILITY: MODES 1 and 2.

ACTIONS

-----NOTE-----  
<DOC A.3> Separate Condition entry is allowed for each control rod scram accumulator.  
-----

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>&lt;3.1.3.5 Act a.1&gt; A. One control rod scram accumulator inoperable with reactor steam dome pressure <math>\geq</math> 900% psig.</p> 	<p>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."</p> <p><u>OR</u></p> <p>A.2 Declare the associated control rod inoperable.</p>	<p>8 hours</p> <p>8 hours</p>

(continued)

Control Rod Scram Accumulators  
3.1.5

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p><i>&lt;3.1.3.5 Act a.2&gt;</i> B. Two or more control rod scram accumulators inoperable with reactor steam dome pressure <math>\geq</math> <del>900</del> psig. 1</p>	<p>B.1 Restore charging water header pressure to <math>\geq</math> <del>1520</del> psig. 940 1</p> <p>AND</p> <p>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</p> <p>OR</p> <p>B.2.2 Declare the associated control rod inoperable. 1 hour</p>	<p>20 minutes from discovery of Condition B concurrent with charging water header pressure <math>&lt;</math> <del>1520</del> psig 940 1</p>
<p><i>&lt;3.1.3.5 Act a.2&gt;</i> C. One or more control rod scram accumulators inoperable with reactor steam dome pressure <math>&lt;</math> <del>900</del> psig. 1</p>	<p>C.1 Verify all control rods associated with inoperable accumulators are fully inserted. AND</p>	<p>Immediately upon discovery of charging water header pressure <math>&lt;</math> <del>1520</del> psig 940 1 (continued)</p>

Control Rod Scram Accumulators  
3.1.5

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>&lt;3.1.3.5 Act a. 2&gt; C. (continued)</p>	<p>C.2 Declare the associated control rod inoperable.</p>	<p>1 hour</p>
<p>&lt;3.1.3.5 Act a. 2, a&gt; D. Required Action and associated Completion Time of Required Action (B.1 or C.1) not met. <span style="border: 1px solid black; padding: 2px;">2</span></p>	<p>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.</p>	<p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>&lt;4.1.3.5&gt; SR 3.1.5.1 Verify each control rod scram accumulator pressure is <math>\geq</math> (1520) psig. <span style="border: 1px solid black; border-radius: 50%; padding: 2px;">940</span> <span style="border: 1px solid black; padding: 2px;">11</span></p>	<p>7 days</p>

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, 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

<DOC M.1>

LCO 3.1.6

OPERABLE control rods shall comply with the requirements of the ~~banked position withdrawal sequence (BPWS)~~.

analyzed rod position sequence

1

<DOC M.1>

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

1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One or more OPERABLE control rods not in compliance with <del>BPWS</del>.</p> <p>the analyzed rod position sequence</p> <p>1</p>	<p>A.1</p> <p>-----NOTE----- Affected control rods may be bypassed in Rod Action Control System (RACS) in accordance with SR 3.3.2.1.8.</p> <p>Move associated control rod(s) to correct position.</p> <p>OR</p> <p>A.2</p> <p>Declare associated control rod(s) inoperable.</p>	<p>Rod Worth Minimizer (RWM)</p> <p>as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation."</p> <p>2</p> <p>8 hours</p> <p>8 hours</p>

<DOC M.1>

(continued)

<CTS>

ACTIONS (continued)

<DOC M.1>

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. Nine or more OPERABLE control rods not in compliance with <del>RPCS</del>. <span style="border: 1px solid black; padding: 2px;">1</span></p> <p><i>analyzed rod position sequence</i></p>	<p>B.1</p> <p>-----NOTE-----  <span style="border: 1px solid black; border-radius: 15px; padding: 2px;">Affected control rods may be bypassed in RACS in accordance with SR 3.3.2.1.8 for insertion only</span></p> <p>Suspend withdrawal of control rods.</p> <p>AND</p> <p>B.2</p> <p>Place the reactor mode switch in the shutdown position.</p>	<p><span style="border: 1px solid black; border-radius: 15px; padding: 2px;">RWM</span> <span style="border: 1px solid black; border-radius: 15px; padding: 2px;">2</span></p> <p><span style="border: 1px solid black; border-radius: 15px; padding: 2px;">as allowed by LCO 3.3.2.1</span></p> <p>Immediately</p> <p>1 hour</p>

SURVEILLANCE REQUIREMENTS

<DOC M.1>

SURVEILLANCE	FREQUENCY
<p>SR 3.1.6.1 Verify all OPERABLE control rods comply with <del>RPCS</del>. <span style="border: 1px solid black; padding: 2px;">1</span></p> <p><i>analyzed rod position sequence</i></p>	<p>24 hours</p>

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1  
ITS: 3.1.6 - ROD PATTERN CONTROL

1. The brackets have been removed and the proper plant specific information/value has been provided.
2. The LaSalle 1 and 2 rod pattern control design does not include a Rod Action Control System, but a rod worth minimizer (RWM), similar to the BWR/4 design. Therefore, the Notes have been modified to reflect the RWM design, and are consistent with the BWR/4 ISTS, NUREG-1433, Revision 1.

<CTS>

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

<LCO 3.1.5> LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

<Appl 3.1.5> 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
<p>&lt;3.1.5 Act a.1&gt;</p> <p>B A I</p> <p>One SLC subsystem inoperable [for reasons other than Condition A].</p>	<p>B.1 A I</p> <p>Restore SLC subsystem to OPERABLE status.</p>	<p>7 days</p> <p>AND</p> <p>10 days from discovery of failure to meet the LCO</p>
<p>&lt;3.1.5 Act a.2&gt;</p> <p>B I</p> <p>Two SLC subsystems inoperable [for reasons other than Condition A].</p>	<p>B.1 B I</p> <p>Restore one SLC subsystem to OPERABLE status.</p>	8 hours
<p>&lt;3.1.5 Act a.1&gt; &lt;3.1.5 Act a.2&gt;</p> <p>D C I</p> <p>Required Action and associated Completion Time not met.</p>	<p>D.1 C I</p> <p>Be in MODE 3.</p>	12 hours

<CTS>

**SURVEILLANCE REQUIREMENTS**

SURVEILLANCE	FREQUENCY
<p>&lt;4.1.5.a.1&gt; SR 3.1.7.1 Verify available volume of sodium pentaborate solution is <del>(<math>\geq</math> 4530 gallons)</del>.  <i>Within the limits of Figure 3.1.7-1</i></p>	<p>24 hours [2]</p>
<p>&lt;4.1.5.a.1&gt; [SR 3.1.7.2 Verify temperature of sodium pentaborate solution is within the limits of <del>Figure 3.1.7-1</del>.  <i>2</i> [2]</p>	<p>24 hours ] [2]</p>
<p>&lt;4.1.5.a.2&gt; [SR 3.1.7.3 Verify temperature of pump suction piping is <del>within the limits of Figure 3.1.7-1</del>.  <i>Up to the storage tank outlet valves</i> <math>\geq 68^{\circ}\text{F}</math> [2]</p>	<p>24 hours ] [2]</p>
<p>&lt;4.1.5.b.2&gt; SR 3.1.7.4 Verify continuity of explosive charge. [3]</p>	<p>31 days</p>
<p>&lt;4.1.5.b.3&gt; SR 3.1.7.5 Verify the concentration of <del>boron</del> in solution is <del>within the limits of Figure 3.1.7-1</del>. [2]</p> <p style="text-align: center;">[4] — Sodium Pentaborate</p>	<p>31 days  AND  Once within 24 hours after water or <del>boron</del> is added to solution  AND  Once within 24 hours after solution temperature is restored within the limits of <del>Figure 3.1.7-1</del>.  [2] — [2]</p>

(continued)

<CTS>

**SURVEILLANCE REQUIREMENTS (continued)**

<4.1.5.6.4>

SR 3.1.7.6 Verify each SLC subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position, or can be aligned to the correct position.

31 days } 2

<4.1.5.c.2>

SR 3.1.7.7 Verify each pump develops a flow rate  $\geq$  41.2 gpm at a discharge pressure  $\geq$  1200 psig.



In accordance with the Inservice Testing Program of 92 days } 2

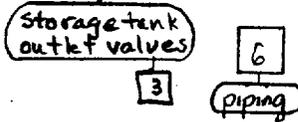
<4.1.5.c.1>

SR 3.1.7.8 Verify flow through one SLC subsystem from pump into reactor pressure vessel.

24 months on a STAGGERED TEST BASIS } 2

<4.1.5.c.4>

SR 3.1.7.9 Verify all heat traced piping between storage tank and pump suction is unblocked.



24 months } 2  
AND  
Once within 24 hours after solution temperature is restored within the limits of Figure 3.1.7-1 } 2

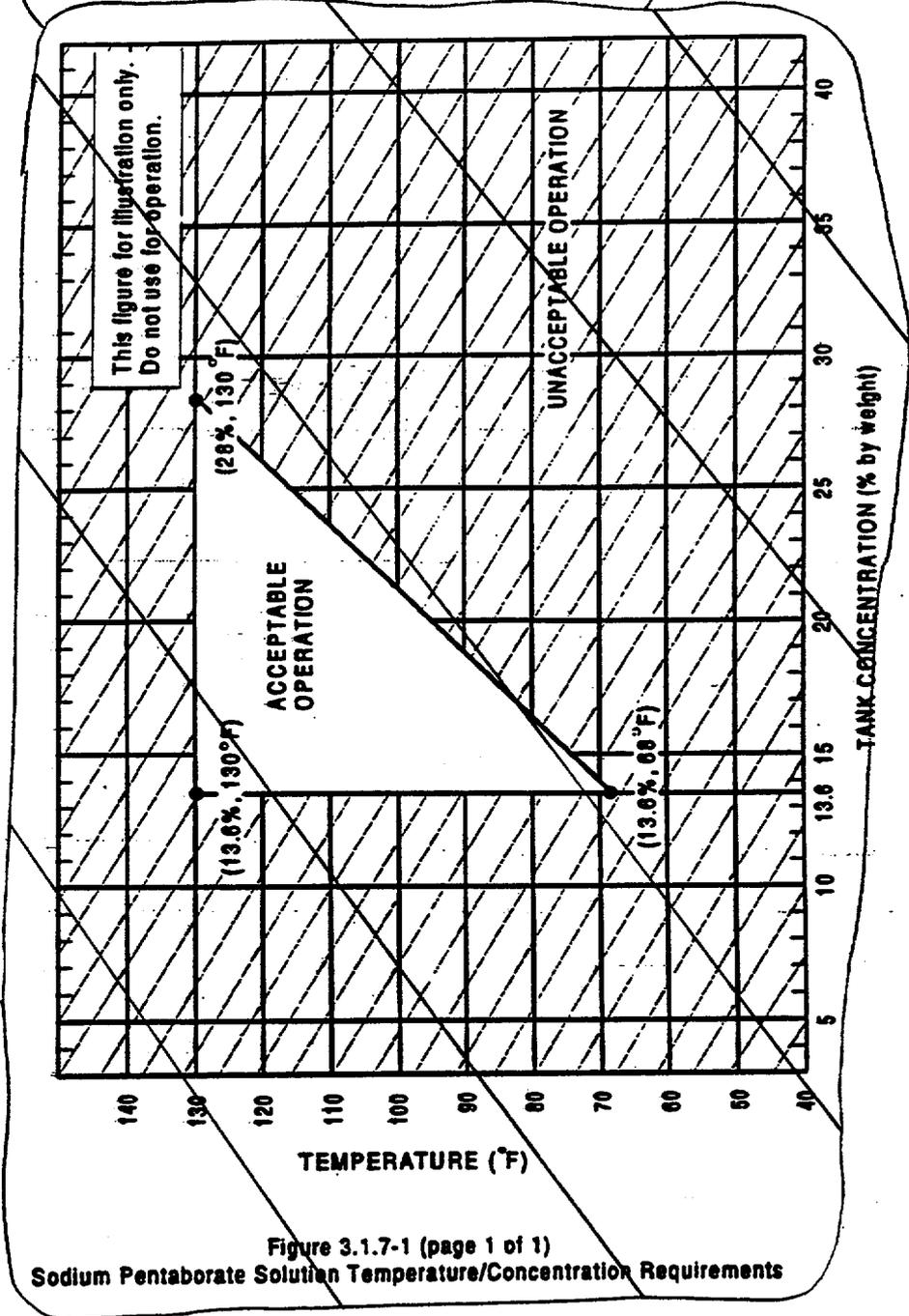
(continued)

**SURVEILLANCE REQUIREMENTS (continued)**

SURVEILLANCE	FREQUENCY
SR 3.1.7.10 Verify sodium pentaborate enrichment is $\geq$ [60.0] atom percent B-10	Prior to addition to SLC tank

5

← insert Figures 3.1.7-1 and 3.1.7-2 → 2



JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1  
ITS: 3.1.7 - STANDBY LIQUID CONTROL SYSTEM

1. The bracketed requirement has been deleted since it is not applicable to LaSalle 1 and 2. The following requirements have been revised and/or renumbered, where applicable, to reflect this deletion.
2. The brackets have been removed and the proper plant specific information/value has been provided.
3. The LaSalle 1 and 2 design is such that heat tracing is only applied up to the storage tank outlet valve of each SLC pump. Therefore, the SR has been changed to reflect this design.
4. The proper LaSalle 1 and 2 nomenclature has been used. This is also consistent with the nomenclature used in SR 3.1.7.1 and SR 3.1.7.2.
5. ISTS SR 3.1.7.10 requires that sodium pentaborate enrichment be verified to be  $\geq [60.0]$  atom percent B-10 prior to addition to the Standby Liquid Control tank. The acceptability of enriched sodium pentaborate is verified, prior to the addition to the Standby Liquid Control tanks at LaSalle 1 and 2, by use of a certificate of compliance provided by the supplier for each batch of enriched sodium pentaborate delivered to LaSalle 1 and 2. The certificate of compliance includes certification that the enrichment of the sodium pentaborate satisfies the acceptance criteria. Therefore, the requirement to perform testing to verify the enrichment of the sodium pentaborate per ISTS SR 3.1.7.10 is not included in the LaSalle 1 and 2 ITS. This change is also consistent with the LaSalle 1 and 2 current licensing basis reflected in Technical Specifications.
6. 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

Doc A.2 LCO 3.1.8 Each SDV vent and drain valve shall be OPERABLE.

Doc A.2 APPLICABILITY: MODES 1 and 2.

ACTIONS

NOTE ⑤ — ①  
Separate Condition entry is allowed for each SDV vent and drain line.

3.1.3.1 fnote #  
3.1.3.1 fnote ##  
3.1.3.1 Act d.1  
3.1.3.1 Act d.1  
3.1.3.1 Act d.2  
3.1.3.1 Act e. 2

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

<CTS>

**SURVEILLANCE REQUIREMENTS**

SURVEILLANCE	FREQUENCY
<p>&lt;4.1.3.1.1.a&gt; 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.</p>	31 days
<p>&lt;4.1.3.1.1.b&gt; SR 3.1.8.2 Cycle each SDV vent and drain valve to the fully closed and fully open position.</p>	92 days
<p>&lt;4.1.3.4&gt; SR 3.1.8.3 Verify each SDV vent and drain valve: a. Closes in <math>\leq</math> <del>30</del><sup>2</sup> seconds after receipt of an actual or simulated scram signal; and b. Opens when the actual or simulated scram signal is reset.</p>	<p>18 months 24 — 2</p>

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1  
ITS: 3.1.8 - SDV VENT AND DRAIN VALVES

1. The ISTS requires that the SDV drain and/or vent valves be restored to operable status if one valve is inoperable. LaSalle 1 and 2 proposes to isolate 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 is consistent with current licensing basis as it has been approved by the NRC in the Safety Evaluation Report for LaSalle Units 1 and 2, Amendments 89 and 74, respectively. In addition, this change is also consistent with the WNP-2 ITS (Amendment 134).
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.

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

<Insert B.3.1.1 ASA>

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 a refueling accident (Ref. 4). 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 3.10.6, "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.

(continued)

Insert B 3.1.1 ASA

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

BASES

APPLICABLE  
SAFETY ANALYSES  
(continued)

Prevention or mitigation of reactivity insertion events is necessary to limit energy deposition in the fuel <sup>positive</sup> thereby preventing significant fuel damage, which could result in undue release of radioactivity. Adequate SDM ensures inadvertent criticalities and potential CRDAs involving high worth control rods (namely the first control rod withdrawal) will not cause significant fuel damage. <sup>10 CFR 50.36(a)(2)(ii)</sup>

2 do

SDM satisfies Criterion 2 of the NRC Policy Statement.

3

2

2

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 <sup>3</sup> during the design process, a design margin is included to account for uncertainties in the design calculations. (Ref. 3)

3

3

2

1

APPLICABILITY

In MODES 1 and 2, SDM must be provided because <sup>to assure shutdown capability</sup> subcriticality with the highest worth control rod withdrawn is assumed in the CRDA analysis (Ref. 3). 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 inadvertent criticality during the withdrawal of a single control rod from a core cell containing one or more fuel assemblies. (Ref. 2)

2

(Ref. 2)

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 6 hour Completion time is acceptable, considering that the reactor can still be shut down.

(continued)

**BASES**

**ACTIONS**

**A.1 (continued)**

assuming no additional failures of control rods to insert, and the low probability of an event occurring during this interval.

**B.1**

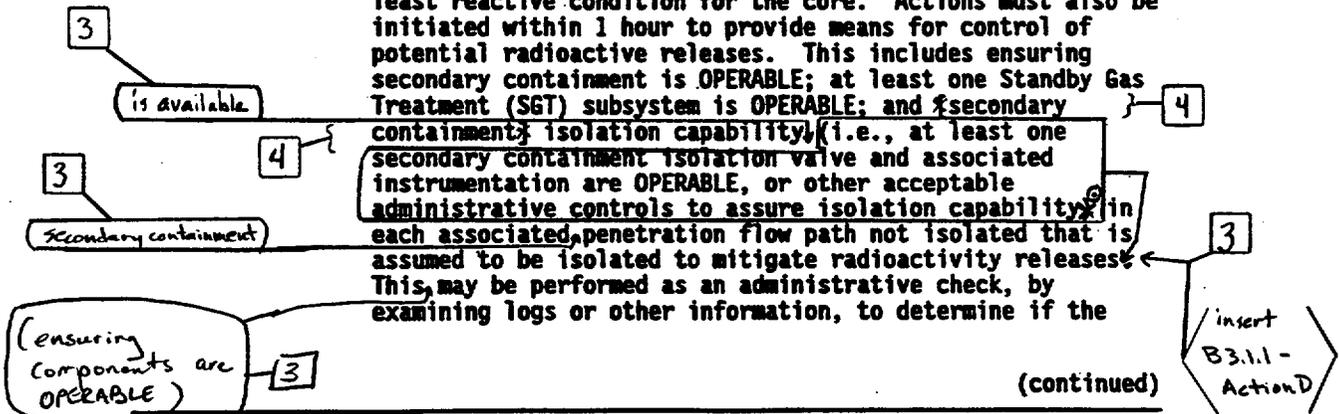
If the SDM cannot be restored, the plant must be brought to MODE 3 within 12 hours, to prevent the potential for further reductions in available SDM (e.g., additional stuck control rods). The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power 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. Actions 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 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



Insert B 3.1.1 Action D

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 B 3.1.1 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

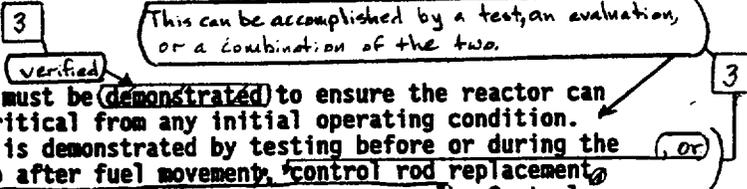
ACTIONS

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

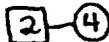
needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, SRs may need to be performed to restore the component to OPERABLE status. Actions must continue until all required components are OPERABLE.

SURVEILLANCE  
REQUIREMENTS

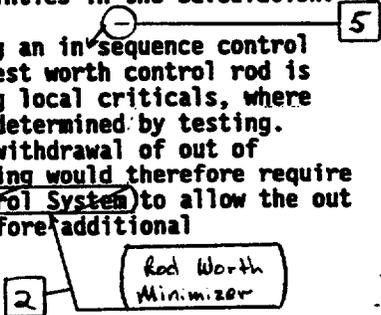
SR 3.1.1.1



Adequate SDM must be ~~demonstrated~~ to ensure the reactor can be made subcritical from any initial operating condition. Adequate SDM is demonstrated by testing before or during the first startup after fuel movement, control rod replacement, or shuffling within the reactor pressure vessel. 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 (i.e., BOC is the most reactive point in the cycle), no correction to the BOC measured value is required (Ref. 6). For the SDM demonstrations that rely solely on calculation of the highest worth control rod, additional margin (0.10%  $\Delta k/k$ ) must be added to the SDM limit of 0.28%  $\Delta k/k$  to account for uncertainties in the calculation.



The SDM may be demonstrated during an in-sequence control rod withdrawal, in which the highest worth control rod is analytically determined, or during local 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 Pattern Control System to allow the out of sequence withdrawal, and therefore additional



(continued)

BASES

SURVEILLANCE  
REQUIREMENTS

SR 3.1.1.1 (continued)

requirements must be met (see LCO 3.10(2), "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 appropriate verification.

During MODES 3 and 4, analytical calculation of SDM may be used to assure the requirements of SR3.1.1.1 are met

3

During MODE 5, adequate SDM is also required to ensure 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 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 or reload sequences inherently satisfy the SR, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM.

REFERENCES

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

2. FSAR, Section [15.4.9].

3. NEDO-21231, "Banked Position Withdrawal Sequence 111," Section 4.1, January 1977.

4. (U) FSAR, Section [15.4.1.1].

5. (U) FSAR, Section [4.3.2.4.1].

6. (U) NDE-24011-P-A-9, "GE Standard Application for Reactor Fuel," Section 3.2.4.1, Sept. 1988.

(as specified in Technical Specification 5.6.5)

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1  
ITS BASES: 3.1.1 - SHUTDOWN MARGIN

1. In the LaSalle reactor safety evaluations, SDM and CRDA analyses are mutually independent. The consideration of SDM is to assure that the reactor is shutdown and remains shutdown with the highest reactivity 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 (highly probable that this will be different from the highest worth control rod determined for SDM) suddenly from the stuck position and falls to the drive position. Doppler reactivity tends to mitigate the event consequences with scram reactivity terminating it.
2. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, or analysis description.
3. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.
4. The brackets have been removed and the proper plant specific information has been provided.
5. Typographical/grammatical error corrected.
6. 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

**BASES**

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**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. 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, 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 ensuring the reactor can be brought safely to cold, subcritical conditions.

1  
Anomalies

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

e.g., gadolinia 2

BACKGROUND  
(continued)

absorbers (if any), control rods, and whatever neutron poisons (mainly xenon and samarium) are present in the fuel.

The predicted core reactivity, as represented by  $k_{eff}$ , 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 monitored  $k_{eff}$  is calculated by the core monitoring system 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  $k_{eff}$  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  $k_{eff}$  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  $k_{eff}$  from the predicted  $k_{eff}$  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 satisfy Criterion 2 of ~~the NRC Policy~~ Statement.

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

2

(continued)

BASES (continued)

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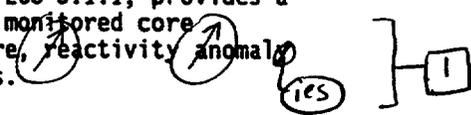
LCO

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 core  $k_{eff}$  and the predicted core  $k_{eff}$  of  $1\% \Delta k/k$  has been established based on engineering judgment. A  $> 1\%$  deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

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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, control rod 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 anomaly is not required during these conditions.



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)

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BASES

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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 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.

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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 further 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

(continued)

BASES

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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 core  $k_{eff}$  values 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.

2

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

REFERENCES

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

2. ~~UFSAR, Chapter 15.3.~~ 3

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JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1  
ITS BASES: 3.1.2 - REACTIVITY ANOMALIES

1. Typographical/grammatical error corrected.
2. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, or analysis description.
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

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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 that under conditions of normal operation, including anticipated operational occurrences, specified acceptable fuel design limits are not exceeded. In addition, the control rods provide the capability to hold the reactor core subcritical under all conditions and to limit the potential amount and rate of reactivity increase caused by a malfunction in the CRD System. The CRD System is designed to satisfy the requirements of GDC 26, GDC 27, GDC 28, and GDC 29, (Ref. 1).

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

1  
and LCO 3.1.6,  
"Rod Pattern  
Control,"

This Specification, along with LCO 3.1.4, "Control Rod Scram Times," and LCO 3.1.5, "Control Rod Scram Accumulators," ensure that the performance of the control rods in the event of a Design Basis Accident (DBA) or transient meets the assumptions used in the safety analyses of References 2, 3, 4, 5, and 6.

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APPLICABLE  
SAFETY ANALYSES

The analytical methods and assumptions used in the evaluations involving control rods are presented in References 2, 3, 4, 5, and 6. The control rods provide the primary means for rapid reactivity control (reactor scram), for maintaining the reactor subcritical, and for limiting

(continued)

BASES

APPLICABLE  
SAFETY ANALYSES  
(continued)

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

The capability of inserting 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 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 that 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 (APLGHGR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), and the fuel damage limit (see Bases for LCO 3.1.6, "Rod Pattern Control") during reactivity insertion events.

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

Control rod OPERABILITY satisfies Criterion 3 of the NRC Policy Statement.

LCO

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

(continued)

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

1  
design

1

**BASES**

LCO  
(continued)

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



**APPLICABILITY**

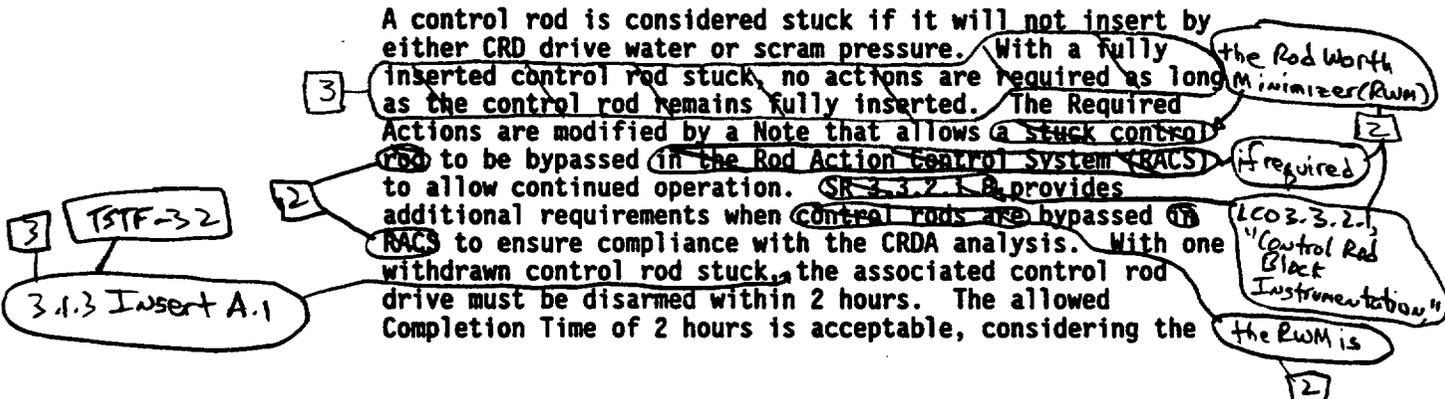
In MODES 1 and 2, the control rods are assumed to function during a DBA or transient and are therefore required to be OPERABLE in these MODES. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is shutdown and a control rod block is applied. This provides adequate 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 that allows a stuck control rod to be bypassed in the Rod Action Control System (RACS) to allow continued operation. SR 3.3.2.1 B provides additional requirements when control rods are bypassed in RACS to ensure compliance with the CRDA analysis. With one withdrawn control rod stuck, the associated control rod drive must be disarmed within 2 hours. The allowed Completion Time of 2 hours is acceptable, considering the



(continued)

### INSERT LCO

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

#### 3.1.3 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 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,

BASES

ACTIONS

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

TSTF-32

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

reactor can still be shut down, assuming no additional control rods fail to insert, and provides a reasonable amount of time 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 by isolating the hydraulic control unit from scram and normal insert and withdraw pressure, yet still maintain cooling water to the CRD.

and normal insert and withdraw pressure

isolation method should also ensure

is maintained

Monitoring of the insertion capability for each withdrawn control rod must also be performed within 24 hours. SR 3.1.3.2 and SR 3.1.3.3 perform periodic tests of the control rod insertion capability of withdrawn control rods. Testing each withdrawn control rod ensures that a generic problem does not exist. The allowed Completion Time of 24 hours provides a reasonable time to test the control rods, considering the potential for a need to reduce power to perform the tests. Required Action A.2 is modified by a

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

Note that states the requirement is not applicable when THERMAL POWER is less than or equal to the actual low power setpoint (LPSP) of the rod pattern controller (RPC), since the notch insertions may not be compatible with the requirements of rod pattern control (LCO 3.1.6) and the RPC (LCO 3.3.2.1, "Control Rod Block Instrumentation")

TSTF-33  
RW M

TSTF-33

TSTF-33

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

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 have failed 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

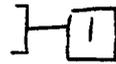
(continued)

BASES

ACTIONS

A.1, A.2, and A.3 <sup>and A.4</sup> TSTF-32 (continued)

postulated additional single failure of an adjacent control rod to insert, sufficient reactivity control remains to reach and maintain MODE 3 conditions (Ref/7).



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 brought to MODE 3 within 12 hours.

must be

~~isolating the control rod from scram prevents damage to the CRDM. The control rod can be isolated from scram by isolating the hydraulic control unit 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 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. Electrically, the control rods can be disarmed by disconnecting power from all four directional control valve solenoids. Required Action C.1 is modified by a Note that allows control rods to be bypassed in the RACS if required to allow insertion of the inoperable control rods and continued operation. ~~SR 3.3.2.1, B~~ provides

the RWM

2

LC03.3.2.1

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

RWM is

2

additional requirements when the control rods are 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

analyzed rod position

2

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

3

in all directions

Out of sequence control rods may increase the potential reactivity worth of a dropped control rod during a CRDA. At  $\leq 10\%$  RTP, the generic banked position withdrawal sequence (BPWS) analysis (Ref 7) 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. A Note has been added to the Condition to clarify 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.

1  
and 8

the analyzed rod position sequence

2

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.

2

(continued)

**BASES**

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**ACTIONS**  
(continued)

**3.1.1**  
If any Required Action and associated Completion Time of Condition A, C, **B**, or **E** are not met or nine or more inoperable control rods exist, 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 (i.e., 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 conditions 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 **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.

**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

(continued)

BASES

SURVEILLANCE  
REQUIREMENTS

SR 3.1.3.2 and SR 3.1.3.3 (continued)

not required when THERMAL POWER is less than or equal to the actual LPSP of the ~~RPC~~ since the notch insertions may not be compatible with the requirements of the ~~Banked Position~~ Withdrawal Sequence (BPWS) (LCO 3.1.6) and the ~~RPC~~ (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.

analyzed rod position sequence

RWM

RWM

3  
Insert  
SR 3.1.3.2 and  
SR 3.1.3.3

SR 3.1.3.4

Verifying the scram time for each control rod to notch position ~~3~~ is  $\leq$  ~~5~~ 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 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

(continued)

INSERT SR 3.1.3.2 AND SR 3.1.3.3

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

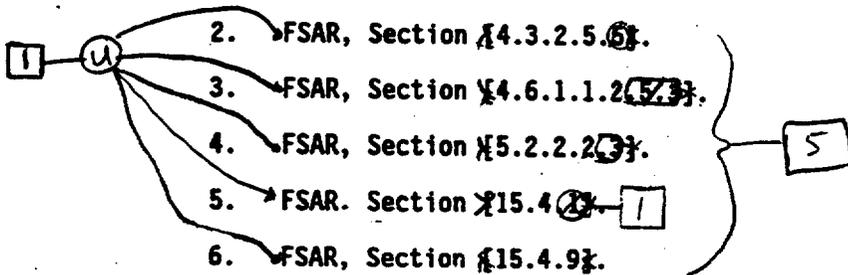
SURVEILLANCE  
REQUIREMENTS

SR 3.1.3.5 (continued)

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 anytime 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. 10 CFR 50, Appendix A, GDC 26, GDC 27, GDC 28, and GDC 29.
2. FSAR, Section ~~4.3.2.5.6~~.
3. FSAR, Section ~~4.6.1.1.2.5.3~~.
4. FSAR, Section ~~5.2.2.2.3~~.
5. FSAR, Section ~~15.4.2~~ 1
6. FSAR, Section ~~15.4.9~~.
7. NEDO-21231, "Banked Position Withdrawal Sequence," Section 7.2, January 1977.



8. NFSR-0091, Commonwealth Edison Topical Report, Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods, (as specified in Technical Specification 5.6.5). 1

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, 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. Changes have been made to reflect those changes made to the Specification. The following requirements have been renumbered, where applicable, to reflect these changes.
3. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.
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

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BACKGROUND

1 Occurrences

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

anticipated

1

When a scram signal is initiated, control air is vented from the scram valves, allowing them to open by spring action. Opening the exhaust valves 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 accumulator pressure drops 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 within 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 References 2, 3, 4, and 5. 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 scrambling slower than the average time, with several control rods scrambling 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.

and 6

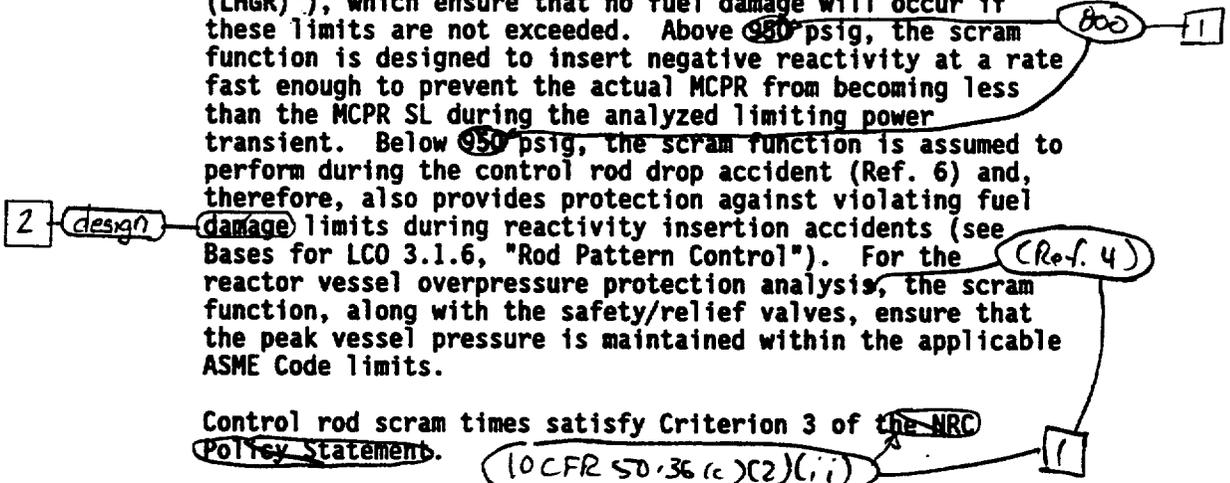
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(continued)

**BASES**

**APPLICABLE SAFETY ANALYSES (continued)**

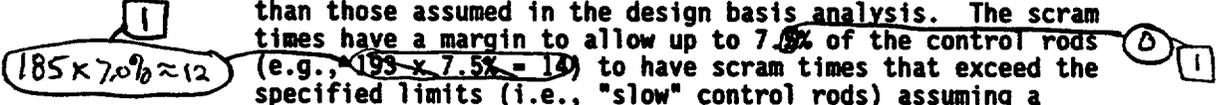
The scram function of the CRD System protects the MCPR Safety Limit (SL) (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), and the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), which ensure that no fuel damage will occur if these limits are not exceeded. Above ~~950~~ 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 ~~950~~ 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.



Control rod scram times satisfy Criterion 3 of the NRC Policy Statement. (10 CFR 50.36 (c) (2) (i))

**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. To account for single failure and "slow" scrambling 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 to allow up to 7.5% of the control rods (e.g., ~~193 x 7.5% = 14~~) to have scram times that exceed 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



(continued)

**BASES**

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LCO  
(continued)

of the specified scram times in Table 3.1.4-1 is accomplished through measurement of the "dropout" times.

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

2  
(face or diagonal)

Table 3.1.4-1 is modified by two Notes, which state control rods with scram times not within the limits of the Table are considered "slow" and that control rods with scram times 3 7 > 3 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 scrambling control rods may be conservatively declared inoperable and not accounted for as "slow" control rods.

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**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."

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**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.

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(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 assumed control rod scram time. Measurement of the scram times with reactor steam dome pressure  $\geq 950$  psig demonstrates acceptable scram times for the transients analyzed in References 3 and 4.

1 Maximum  
of approximately  
800 psig  
4 800

Scram insertion times increase with increasing reactor pressure 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  $\geq 950$  psig ensures that the 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 scram time testing is performed within a reasonable time following fuel movement within the reactor pressure vessel or after a shutdown  $\geq 120$  days, control rods are required to be tested before exceeding 40% RTP. 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  $\geq 120$  days, all control rods are required to be scram time tested. This frequency is acceptable, considering the additional surveillances performed for control rod OPERABILITY, the frequent verification of adequate accumulator pressure, and the required testing of control rods affected by work on control rods or the CRD System.

fuel movement within the associated core cell and by

TSTF-222

(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." If more than 20% of the sample is 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 were 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."

5 (i.e.)

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 that the affected control rod is still within acceptable limits. The limits for reactor pressures < 950 psig are established based on a high probability of meeting the acceptance criteria at reactor pressures  $\geq$  950 psig. Limits for reactor pressures  $\geq$  950 psig are found in Table 3.1.4-1. If testing demonstrates the affected control rod does not meet these limits, but is within 7-second limit of Table 3.1.4-1, Note 2, the control rod can be declared OPERABLE and "slow."

2  
found in the  
Technical  
Requirements  
Manual (Ref. 7)  
and are

2  
scram time  
4  
800

(continued)

**BASES**

**SURVEILLANCE  
REQUIREMENTS**

SR 3.1.4.3 (continued)

Specific examples of work that could affect the scram times include (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 valves in the piping required for scram.

The Frequency of once prior to declaring the affected control rod OPERABLE is acceptable because of the capability of testing 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,

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  $\geq$  950 psig. Where work has been performed at high reactor pressure, the requirements of SR 3.1.4.3 and SR 3.1.4.4 will be satisfied with one test. For a control rod affected by work performed while shut down, however, a zero pressure and a high pressure test may be required. This testing ensures that the control rod scram performance is acceptable for operating reactor pressure conditions prior to withdrawing the control rod for continued operation. Alternatively, a test during hydrostatic pressure testing could also satisfy both criteria.

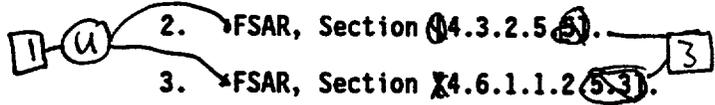
The Frequency of once prior to exceeding 40% RTP is acceptable because of the capability of testing the control rod at the different conditions and the more frequent surveillances on other aspects of control rod OPERABILITY.

TSTF-222  
4  
800  
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.

TSTF-222

**REFERENCES**

1. 10 CFR 50, Appendix A, GDC 10.
2. FSAR, Section 4.3.2.5 (1).
3. FSAR, Section 4.6.1.1.2 (5.3).



(continued)

**BASES**

**REFERENCES**  
(continued)

- 4. FSAR, Section ~~5.2.2~~ 2.2.3
  - 5. FSAR, Section ~~15.4~~ 1
  - 6. FSAR, Section ~~15.4.9~~
  - 7. *Technical Requirements Manual.* 1
- 
- Handwritten annotations: A box containing the number '1' and a circle containing the number '4' are connected by lines to items 4, 5, and 6. A bracket groups items 4, 5, and 6, pointing to a box containing the number '3'. Item 7 has a box containing the number '1' next to it.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, 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, or analysis 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/value has been provided.
4. Changes have been made to reflect those changes made to the Specification.
5. Typographical/grammatical error corrected.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Control Rod Scram Accumulators

BASES

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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."

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APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the control rod scram function are presented in References 1, 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. 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.

SL 2.1.1,  
"Reactor Core  
SLs" and  
11

The scram function of the CRD System, and, therefore, the OPERABILITY of the accumulators, protects the MCPR Safety Limit (see Bases for LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), which ensure that no fuel damage will occur if these limits are not exceeded (see Bases for LCO 3.1.4). Also, 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 the NRC Policy Statement.

10CFR 50.36(c)(2)(i) - 2

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~~ 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 under these conditions. Requirements for scram accumulators in MODE 5 are contained in LCO 3.9.5, "Control Rod OPERABILITY - Refueling."

ACTIONS

The ACTIONS <sup>44</sup> 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 action for each affected control rod. Complying with the Required Actions may allow for continued operation and subsequent affected control rods governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

With one control rod scram accumulator inoperable and the reactor steam dome pressure  $\geq 900$  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, which clarifies that declaring the control rod "slow" is only applicable if the associated

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

**4** Rod **3** Surveillance  
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 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 considered 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

**5** 940  
With two or more control rod scram accumulators inoperable and reactor steam dome pressure  $\geq 900$  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  $< 1520$  psig concurrent with Condition B, adequate charging water header pressure must be restored. The allowed Completion Time of 20 minutes is considered a reasonable time to place a CRD pump into service to restore the charging header pressure, if required. This Completion Time also recognizes the ability of the reactor pressure alone to fully insert all control rods.

**4** Rod **3** Surveillance  
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" is only applicable 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

(continued)

BASES

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

May  
4

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 considered 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 < 1520 psig, concurrent with Condition C, all control rods associated with inoperable accumulators must be verified to be fully inserted. Withdrawn control rods with inoperable scram 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 the accumulator is inoperable.

6

940

D.1

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

(continued)

**BASES**

**ACTIONS**

D.1 (continued)

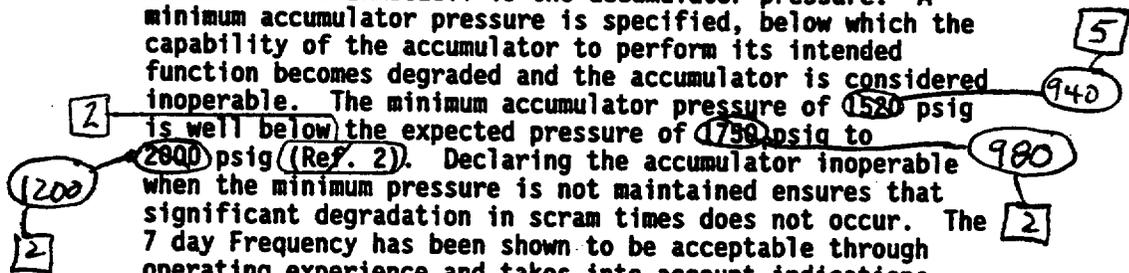
Completion Time associated with loss of the CRD charging pump (Required Actions B.1 and C.1) cannot be met. This ensures that all insertable control rods are inserted and that the reactor is in a condition that does not require the active function (i.e., scram) of the control rods. This Required Action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

2

**SURVEILLANCE REQUIREMENTS**

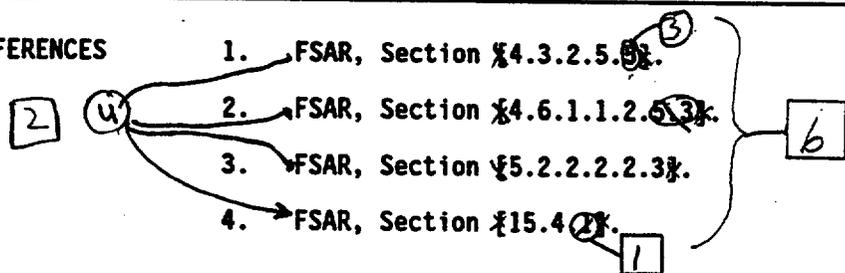
SR 3.1.5.1

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 1520 psig is well below the expected pressure of 1750 psig to 2000 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. FSAR, Section §4.3.2.5.3.
2. FSAR, Section §4.6.1.1.2.5.3.
3. FSAR, Section §5.2.2.2.2.3.
4. FSAR, Section §15.4.2.



JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1  
ITS BASES: 3.1.5 - CONTROL ROD SCRAM ACCUMULATORS

1. The limits protected by the control rod scram function (and thus the accumulators) are also discussed in the Bases for SL 2.1.1, "Reactor Core SLs." This addition was approved in BWR-18, C.13, but was apparently inadvertently left out of Revision 1 to this NUREG (it was included in the BWR ISTS, NUREG-1433, Revision 1).
2. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, or analysis description.
3. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.
4. Typographical/grammatical error corrected.
5. Changes have been made to reflect those changes made to the Specification. The following requirements have been renumbered, where applicable, to reflect these changes.
6. The brackets have been removed and the proper plant specific information/value has been provided.
7. Changes have been made to more closely reflect the requirements of the Specification.

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 pattern controller (RPC) (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 effectively limit the potential amount of reactivity addition that could occur in the event of a control rod drop accident (CRDA). *Worth Minimizer*

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, and 3. CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analysis. The RPC (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 undue release of radioactivity. Since the failure consequences for UO<sub>2</sub> 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. 5 and 6). Generic evaluations (Refs. 7 and 8) 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. 9) and the calculated offsite doses will be well within the required limits (Ref. 10).

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

All changes are [3] unless otherwise indicated

Rod Pattern Control  
B 3.1.6

BASES

predetermined sequencing rules  
(analyzed rod position sequence)

the cycle specific analyses

APPLICABLE SAFETY ANALYSES (continued)

Control rod patterns analyzed in Reference 1 follow the banked position withdrawal sequence (BPWS) described in Reference 2. 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 defined to minimize the maximum incremental control rod worths without being overly restrictive during normal plant operation. The generic BPWS analysis (Ref. 3) 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.

analyzed rod position sequence

5

1, 2, 3, 4, and 5

Cycle specific analyses ensure that the 280 cal/gm fuel design limit will not be violated during a CRDA under worst case scenarios.

Cycle specific analyses

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

Rod pattern control satisfies the requirements of 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  $\leq$  10% RTP, the CRDA is a Design Basis Accident (DBA) 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.

4 and 5

design

6

16

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.

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

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

2

4

1 the Run

Required Action A.1 is modified by a Note, which allows ~~control rods~~ to be bypassed ~~in Rod Action Control System (RACS)~~ to allow the affected control rods to be returned to their correct position. 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.

1

helps to 6

OPERABILITY of control rods is determined by compliance with LCO 3.1.3; 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.

5

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)

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

(continued)

**BASES** 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)

Rod Pattern Control  
B 3.1.6

**ACTIONS** 1) B.1 and B.2 (continued) **RWM** that allows the affected control rods to be bypassed in RACS in accordance with SR 3.1.6.1 to allow insertion only.

3) analyzed rod position sequence

With nine or more OPERABLE control rods not in compliance with **RPM**, the reactor mode switch must be placed in the shutdown position within 1 hour. With the reactor mode switch in shutdown, the reactor is shut down, and therefore 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 analyzed rod position sequence 3)

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

**REFERENCES**

3) INSERT REF-1

1. Current Cycle Safety Analysis.
2. "Modifications to the Requirements for Control Rod Drop Accident Mitigating Systems," BWR Owners Group, July 1987.
3. FSAR, Section 15.4.9.

6) 7) NUREG-0979, "NRC Safety Evaluation Report for GESSAR II BWR/6 Nuclear Island Design, Docket No. 50-447," Section 4.2.1.3.2, April 1983.

7) 8) NUREG-0800, "Standard Review Plan," Section 15.4.9, "Radiological Consequences of Control Rod Drop Accident (BWR)," Revision 2, July 1981.

(continued)

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).

BASES

REFERENCES  
(continued)

⑪⑫ 10 CFR 100.11, "Determination of Exclusion Area Low Population Zone and Population Center Distance."

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

⑩⑪⑫ ASME, Boiler and Pressure Vessel Code.

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

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

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1  
ITS BASES: 3.1.6 - ROD PATTERN CONTROL

1. Changes have been made to reflect those changes made to the Specification.
2. The brackets have been removed and the proper plant specific information/value has been provided.
3. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, or analysis description.
4. 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.
5. 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.
6. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

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 (ATWS).

The SLC System consists of a boron solution storage tank, two positive displacement pumps, two explosive valves, which 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 (through the high pressure

core/spray system sparger). 1  
*near the bottom of the core shroud, where it mixes with the cooling water rising through the core*

**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 not enough control rods can be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to compensate for all of the various reactivity effects that could occur during plant operation. To meet this objective, it is necessary to inject a quantity of boron that produces a concentration of 660 ppm of natural boron in the reactor core at 68°F. To allow for potential leakage and imperfect mixing in the reactor system, an additional amount of boron equal to 25% of the amount cited above is added (Ref. 2). The (temperature versus volume) concentration limits in Figure 3.1.7-1 are calculated such that the required concentration is achieved, accounting for dilution in the RPV with normal water level and including the water volume in 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

1  
a reactivity change equivalent to

1  
An additional 250 ppm is provided to accommodate

1 by

2  
determines

1  
enriched  
Insert B3.1.7 ASA

1

2

(continued)

Insert B 3.1.7 ASA

To ensure this objective is met, a sodium pentaborate solution enriched with boron-10 is used. The shutdown analysis assumes a sodium pentaborate solution with enriched boron is used (Ref. 2). A 45% enriched sodium pentaborate solution is also used to satisfy the requirements of Reference 1.

**BASES**

**APPLICABLE  
SAFETY ANALYSES  
(continued)**

tank. No credit is taken for the portion of the tank volume that cannot be injected.

The SLC System satisfies <sup>Criterion 4</sup> ~~(the requirements)~~ <sup>10CFR 50.36(c)(2)(ii)</sup> of the NRC Policy Statement because operating experience and probabilistic risk assessment have generally shown it to be important to public health and safety.

3

**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 containing an OPERABLE pump, an explosive valve and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path.

**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 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 during these conditions, when only a single control rod can be withdrawn.

**ACTIONS**

**A.1**

4

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

(continued)

BASES

ACTIONS

A.1 (continued)

function. Because of the low probability of an ATWS event and 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 to 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.

4

4

A

B.1

If one SLC System 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 function. However, the overall reliability is reduced because a single failure in the remaining OPERABLE subsystem could result in reduced SLC System shutdown capability. The 7 day Completion Time is based on the availability of an OPERABLE subsystem capable of performing the intended SLC System function and the low probability of a Design Basis Accident (DBA) or severe transient occurring

unit shutdown

and inability to meet the requirements of Reference 1.

(continued)

BASES

ACTIONS

A → 0.1 (continued)

4

concurrent with the failure of the Control Rod Drive System to shut down the ~~plant~~ reactor

1

4

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 of 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.

4

B → 0.1

4

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.

4

C → 0.1

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

(continued)

BASES

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ACTIONS

④ (C) → ④.1 (continued)

experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

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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 the SLC System OPERABILITY without disturbing normal plant operation. These Surveillances ensure the proper borated solution 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 24 hour Frequency of these SRs is based on operating experience that has shown there are relatively slow variations in the measured parameters of volume and temperature.

①  
(using the local indicator)  
up to the storage tank outlet valves  
④

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 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 that has demonstrated the reliability of the explosive charge continuity.

SR 3.1.7.6 verifies each valve in the system is in its correct position, but does not apply to the squib (i.e., explosive) valves. Verifying the correct alignment for manual, power operated, and automatic valves in the SLC System flow path ensures 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

(continued)

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BASES

SURVEILLANCE  
REQUIREMENTS

SR 3.1.7.4 and SR 3.1.7.6 (continued)

since the SLC System is a manually initiated system. This Surveillance does not apply to valves that are locked, sealed, or otherwise secured in position, since they were verified to be in the correct position prior to locking, sealing, or securing. This verification of valve alignment does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct positions. The 31 day Frequency is based on engineering judgment and is consistent with the procedural controls governing valve operation that ensure correct valve positions.

SR 3.1.7.5

Sodium pentaborate

This Surveillance requires an examination of the sodium pentaborate solution by using chemical analysis to ensure 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 establish that the ~~boron~~ solution concentration is within the specified limits. This Surveillance must be performed anytime the temperature is restored to within the limits of Figure 3.1.7-1, to ensure 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.

4

Sodium pentaborate

4

SR 3.1.7.7

Demonstrating each SLC System pump develops a flow rate  $\geq 41.2$  gpm at a discharge pressure  $\geq$  ~~(1200)~~ 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

1220

4

(continued)

BASES

SURVEILLANCE  
REQUIREMENTS

SR 3.1.7.7 (continued)

5 tests → inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this Surveillance is in accordance with the Inservice Testing Program (IP) 92 days.

6

7

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 months, at alternating 18 month intervals. The Surveillance may be performed in separate steps to prevent injecting boron into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a test tank through one SLC subsystem and into the RPV. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance test when performed at the 18 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

1 verify flow

4 24

in the flow path

1 storage tank outlet valves

4

1 up to the storage tank outlet valves

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 tank. The 18 month Frequency is acceptable since there is a low probability that the subject piping will be blocked due to precipitation of the boron from solution in the heat traced piping. This is especially true in light of the daily temperature verification of this piping required by SR 3.1.7.3. However, if, in performing SR 3.1.7.3, it is

(continued)

Upon completion of this verification, the pump suction piping between the storage tank outlet valves and pump suction must be drained and flushed with demineralized water, since the piping is not heat traced.

1

**BASES**

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**SURVEILLANCE  
REQUIREMENTS**

SR 3.1.7.8 and SR 3.1.7.9 (continued)

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 within the limits of Figure 3.1.7-4.



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.

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**REFERENCES**

1. 10 CFR 50.62.



2. u PSAR, Section {9.3.5.3}.



JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, 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, or analysis description.
2. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.
3. 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.
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. Typographical/grammatical error corrected.
6. The IST program for LaSalle 1 and 2 is not required to provide information for trend purposes.
7. The brackets have been removed and the proper plant specific information/value has been provided.

**B 3.1 REACTIVITY CONTROL SYSTEMS**

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

**BASES**

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**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 consists of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two headers and two instrument volumes, each receiving approximately one half of the control rod drive (CRD) discharges. The two instrument volumes are connected to a common drain line with two valves in series. Each header is connected to a common vent line with 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.

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**APPLICABLE SAFETY ANALYSES**

The Design Basis Accident and transient analyses assume all the control rods are capable of scrambling. The primary function of the SDV is to limit the amount of reactor coolant discharged during a scram. 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 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 also

(continued)

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BASES

APPLICABLE  
SAFETY ANALYSES  
(continued)

allow continuous drainage of the SDV during normal plant operation to ensure 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 exceeds a specified setpoint. The setpoint is chosen such 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)



LCO

The OPERABILITY of all SDV vent and drain valves ensures that, during a scram, the SDV vent and drain valves will close 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 be open to ensure that a path is available for the SDV piping to drain freely at other times.

APPLICABILITY

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

2

2

ACTIONS

The ACTIONS <sup>3</sup>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

(continued)

BASES

ACTIONS  
(continued)

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

the line must be isolated to contain the reactor coolant during a scram 4

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

7 day

and the lines) not isolated 4

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 Note that allows periodic draining and venting of the SDV when a line is isolated.~~

4

During these periods, the line may be unisolated under administrative control. This allows any accumulated water in the line to be drained, to preclude a reactor scram on SDV high level. This is acceptable, since the administrative controls ensure the valve can be closed quickly, by a dedicated operator, if a scram occurs with the valve open. at the valve controls

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.

3

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)

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

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ACTIONS

C.1 (continued) 3

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.

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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 function 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

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

<01 34>

2  
"Control Rod  
OPERABILITY,"

SR 3.1.8.3 (continued)

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

24  
4

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REFERENCES

1. FSAR, Section ~~4.6.1.1.2~~ ~~2.6~~ 5
2. 10 CFR 100.
3. NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," August 1981.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, 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, or analysis 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.

**GENERIC NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: SECTION 3.1 - REACTIVITY CONTROL SYSTEMS**

**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.

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: SECTION 3.1 - REACTIVITY CONTROL SYSTEMS

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 NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: SECTION 3.1 - REACTIVITY CONTROL SYSTEMS

**"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 NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: SECTION 3.1 - REACTIVITY CONTROL SYSTEMS

"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-1434, 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 NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: SECTION 3.1 - REACTIVITY CONTROL SYSTEMS

**"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 tests 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 NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: SECTION 3.1 - REACTIVITY CONTROL SYSTEMS

"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.

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: SECTION 3.1 - REACTIVITY CONTROL SYSTEMS

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 LaSalle 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.

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: SECTION 3.1 - REACTIVITY CONTROL SYSTEMS

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-1434, 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.

NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: 3.1.1 - SHUTDOWN MARGIN

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: 3.1.1 - SHUTDOWN MARGIN

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 deletes a Surveillance Requirement that provides only confirmatory information. The change would not affect equipment design or operation and involves only a Surveillance of a specified parameter which is not considered as an accident initiator. Therefore, the deletion of this Surveillance will not significantly increase the probability of an accident previously evaluated. Further, since the Surveillance would provide only confirmation of the parameter value for which sufficient uncertainties and biases have been adequately considered in the limit development, the deletion of the Surveillance 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. The Surveillance provides only confirmation of an adequately known value of a parameter for which sufficient uncertainties and biases have been adequately considered in the limit development. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The SDM limits account for uncertainties and biases, and for fuel cycle changes. The required margin is determined by the initial startup test and as corroborated by the periodic reactivity anomaly Surveillance (CTS 4.1.2). Therefore, the additional Surveillance Requirement provides no additional useful information and 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.

NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: 3.1.2 - REACTIVITY ANOMALIES

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.

NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: 3.1.2 - REACTIVITY ANOMALIES

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  $k_{\text{eff}}$  and the predicted  $k_{\text{eff}}$ . 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.

NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: 3.1.3 - CONTROL ROD OPERABILITY

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.

NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: 3.1.3 - CONTROL ROD OPERABILITY

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 allows unlimited continued operation with one stuck control rod. Operation with one stuck control rod is not assumed in the initiation of any accidents previously evaluated, and therefore, cannot increase the probability of such accidents. Additionally, the current analysis is insensitive to one inoperable fully withdrawn rod and one additional single failure. 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 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. The proposed change does not involve a significant reduction in a margin of safety, since prompt, regular surveillances will confirm that no more than one control is stuck and the SDM is maintained.

NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: 3.1.3 - CONTROL ROD OPERABILITY

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 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.

NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: 3.1.3 - CONTROL ROD OPERABILITY

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 extends the Surveillance Frequency for partially withdrawn control rods. The change does not affect equipment design or operation. The affected Surveillance is not considered to be an accident initiator. Therefore, this change will not significantly increase the probability of an accident previously evaluated. Furthermore, extension of the Surveillance Frequency will not impact the ability of the system to perform its function following an accident. Therefore, the 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 extension 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 provides a minor reduction in the probability of finding an inoperable control rod. Most of the control rods will continue to be tested on the current Frequency. However, if one stuck rod is identified, all rods must be checked promptly. 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.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 replaces the daily control rod notch test with one performed once within 24 hours. The intent of the current daily test of control rods is to ensure that a generic problem does not exist and that control rod insertion capability remains. The proposed single performance within 24 hours provides the information to be used in determining whether a generic problem exists and control rod insertion capability remains.

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

2. 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 modifications to the plant.

3. Does this change involve a significant reduction in a margin of safety?

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

NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: 3.1.3 - CONTROL ROD OPERABILITY

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 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.

NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: 3.1.3 - CONTROL ROD OPERABILITY

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 deletes the requirement for increased frequency of control rod testing when three or more 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.

NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: 3.1.3 - CONTROL ROD OPERABILITY

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 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.

NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: 3.1.3 - CONTROL ROD OPERABILITY

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 proposed change would increase the time allowed to accomplish recoupling, allow bypassing of the RWM to recouple, and remove the restriction for a single attempt 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.

NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: 3.1.3 - CONTROL ROD OPERABILITY

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 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 acceptable 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 acceptable 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.

NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: 3.1.3 - CONTROL ROD OPERABILITY

L.12 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.

NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: 3.1.3 - CONTROL ROD OPERABILITY

L.13 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 verify no rod drift alarm every 12 hours if the control rod is at a position without an operable position indication does not need to be explicitly stated in the Technical Specifications. The control rod position test required by SR 3.1.3.1 ensures the position is known every 24 hours. If the position is not known, then 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 verify no rod drift alarm every 12 hours if the control rod is at a position without an operable position indication does not impact any margin of safety. The control rod position test required by SR 3.1.3.1 ensures the position indication must be available and is performed every 24 hours. If the position is not known, the appropriate actions must be taken for inoperable control rods in accordance with the ACTIONS of ITS 3.1.3. In addition, position of the control rods is indicated in the control room, as well as the rod drift alarm. Operators would be aware if a rod drifted. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: 3.1.3 - CONTROL ROD OPERABILITY

L.14 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 and control rod coupling verification surveillances, and by verifying full-in position indicators prior to startup and when a control rod is fully inserted 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 and control rod coupling verifications required by SR 3.1.3.5, 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 and control rod coupling verification surveillances, and by verifying full-in position indicators prior to startup and when a control rod is fully inserted 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 and control rod coupling verifications required by SR 3.1.3.5, 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

NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: 3.1.3 - CONTROL ROD OPERABILITY

L.14 CHANGE

3. (continued)

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 (except for normal control rod movement) 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 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

NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: 3.1.4 - CONTROL ROD SCRAM TIMES

L.1 CHANGE

3. (continued)

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.

NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: 3.1.5 - CONTROL ROD SCRAM ACCUMULATORS

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.

NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: 3.1.7 - STANDBY LIQUID CONTROL SYSTEM

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 deletes the requirements for Standby Liquid Control (SLC) System OPERABILITY during refueling. The SLC System is not assumed to initiate any previously evaluated events and, therefore, the proposed change will not affect the probability of a previously analyzed accident. The SLC System is not assumed to operate in the mitigation of any previously analyzed accident assumed to occur during refueling. Therefore, the proposed 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 removes a backup to the available systems for reactivity control during plant shutdown conditions. However, this backup is not considered in the margin of safety when determining the required reactivity for refueling events. Therefore, the proposed change does not allow operations which would involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: 3.1.7 - STANDBY LIQUID CONTROL SYSTEM

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 current monthly requirement to start both SLC pumps and recirculate demineralized water to the test tank is proposed to be deleted. There is no reason that the pumps should require testing every 30 days based on previous surveillance history. The testing required by the Inservice Test Program, 10 CFR 50.55a, and proposed SR 3.1.7.7 provides adequate assurance that the pumps will perform their required function. Eliminating the more frequent monthly testing, based on the performance history of the SLC pumps and the continued testing in accordance with the Inservice Testing Program, 10 CFR 50.55a, and SR 3.1.7.7, will ensure the reliability of the SLC pumps to perform their required function. Therefore, since reliability of the pumps will not be affected, 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 change only involves elimination of unnecessary testing of the SLC pumps. The reliability of the pumps will be maintained in accordance with the Inservice Testing Program, 10 CFR 50.55a, and proposed SR 3.1.7.7. 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 change eliminates unnecessary testing of the SLC pumps. The reliability of the pumps will be maintained through testing in accordance with the Inservice Testing Program, 10 CFR 50.55a, and proposed SR 3.1.7.7. In that the reliability of the pumps will not be affected by elimination of the monthly testing, no significant reduction in a margin of safety is involved with this change.

NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: 3.1.7 - STANDBY LIQUID CONTROL SYSTEM

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?

This change would remove a specific restriction to perform Surveillances of the Standby Liquid Control (SLC) System during shutdown. The SLC System is not considered as an initiator of any previously analyzed accident. Therefore, this change does not significantly increase the frequency of such accidents. The appropriate plant conditions for performance of the Surveillance will continue to be controlled to assure the potential consequences are not significantly increased. This control method has been previously determined to be acceptable as indicated in Generic Letter 91-04. Therefore, this change does not significantly increase the consequences of any previously analyzed accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

This change removes a specific restriction on the plant conditions for performing Surveillances but does not change the method of performance. The appropriate plant conditions for performance of the Surveillances will continue to be controlled to assure the possibility for a new or different kind of accident are not created. This control method has been previously determined to be acceptable as indicated in Generic Letter 91-04. Therefore, this change does not create the possibility of a new or different kind of accident from any previously analyzed accident.

3. Does this change involve a significant reduction in a margin of safety?

The margin of safety considered in determining the appropriate plant conditions for performing the Surveillances will continue to be controlled to assure that there is no significant reduction. This control method has been previously determined to be acceptable as indicated in Generic Letter 91-04. Therefore, the change does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: 3.1.7 - STANDBY LIQUID CONTROL SYSTEM

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 would delete a redundant portion of a specific Surveillance. The remaining Surveillance of the SLC System provides sufficient assurance that the system will operate as designed. The SLC System is not assumed to initiate any previously evaluated events and therefore the proposed change will not affect the probability of a previously analyzed accident. The deletion of the Surveillance does not affect the capability of the system to operate in the mitigation of any previously analyzed accidents since another Surveillance provides assurance of that capability. Therefore, the proposed 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 or a new mode of operation 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?

The proposed change would remove a backup verification of an alternate method to determine the availability of the system. However, this backup is not considered in the margin of safety when determining the required reactivity for refueling events. Therefore, the proposed change does not allow operations which would involve a significant reduction in the margin of safety.

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: 3/4.1.3.8 - 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 3.65 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.

NO SIGNIFICANT HAZARDS CONSIDERATION  
CTS: 3/4.1.3.8 - CONTROL ROD DRIVE HOUSING SUPPORT

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: 3/4.1.6 - 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)

LC0 3.2.2 All MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR.

APPLICABILITY: THERMAL POWER  $\geq$  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 $\geq$ 25% RTP  <u>AND</u> 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  <u>AND</u>  Once within 72 hours after each completion of SR 3.1.4.2  <u>AND</u>  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  <u>AND</u> 24 hours thereafter

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

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**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 criteria specified in 10 CFR 50.46 are met during the postulated design basis loss of coolant accident (LOCA). Additionally, for GE 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).

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**APPLICABLE SAFETY ANALYSES**

The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs) that determine APLHGR limits are presented in UFSAR, Chapters 4, 6, and 15, and in References 1 and 2.

LOCA analyses are performed to ensure that the 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 2. 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. This additional limitation is due to the conservative analysis assumption of an earlier

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BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

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 the UFSAR, Chapters 4, 6, and 15, and in Reference 1. 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 A00s for operation with LHGR 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 A00s.

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

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LCO

The APLHGR limits specified in the COLR are the result of 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.

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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 (IRM) scram function and the average power range monitor (APRM) scram function provide prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in

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BASES

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APPLICABILITY      MODE 2. Therefore, at THERMAL POWER levels  $\leq$  25% RTP, the  
(continued)      reactor operates with margin to the APLHGR limits; thus,  
this LCO is not required.

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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 is taken to restore the APLHGR(s) to within the required limits such that the plant will be operating within analyzed conditions and within the 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.

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SURVEILLANCE  
REQUIREMENTS

SR 3.2.1.1

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

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(continued)

BASES (continued)

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- REFERENCES
1. NEDO-24011-P-A, "General Electric Standard Application for Reactor Fuel," (as specified in Technical Specification 5.6.5).
  2. EMF-94-217(NP), Revision 1, "Boiling Water Reactor Licensing Methodology Summary," November 1995.
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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

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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 (A00s). Although fuel damage does not necessarily occur if a fuel rod actually experiences 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.

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APPLICABLE  
SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the A00s 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 ( $\Delta$ CPR). When the largest  $\Delta$ CPR is added to the MCPR SL, the required operating limit MCPR is obtained.

(continued)

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BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state (MCPR<sub>f</sub> and MCPR<sub>p</sub>, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency as identified in the 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 the 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 power/flow state is the greater of the rated conditions MCPR operating limit or the power 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.

Power dependent MCPR limits (MCPR<sub>p</sub>) are determined on a cycle-specific basis. These limits are established to protect the core from plant transients other than core flow increases, including pressurization and local control rod withdrawal events.

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

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LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. MCPR operating limits which include the effects of analyzed equipment out-of-service are also included in the COLR. The MCPR operating limits are determined by the larger of the MCPR<sub>f</sub> and MCPR<sub>p</sub> limits.

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(continued)

BASES (continued)

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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 slow recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs.

Studies of the variation of limiting transient behavior have been performed over the range of power and flow conditions. These studies (Ref. 5) 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 (IRM) and average power range monitor (APRM) provide rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < 25% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required.

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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 the 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

(continued)

BASES

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ACTIONS

B.1 (continued)

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.

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SURVEILLANCE  
REQUIREMENTS

SR 3.2.2.1

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

SR 3.2.2.2

Because the transient analyses may take credit for conservatism in the control rod 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 either on the applicable limit associated with scram times of LCO 3.1.4, "Control Rod Scram Times," or the realistic scram times. The scram time dependent MCPR limits are contained in the COLR. This determination must be performed within 72 hours after each set of control rod 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 control rod scram speed distribution expected during the fuel cycle.

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(continued)

BASES (continued)

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- 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).
  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-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).
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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

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**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 the LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences (A00s). 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.

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**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<sub>2</sub> 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 A00 limits, plus an allowance for densification power spiking.

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

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(continued)

BASES (continued)

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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.

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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 margin to the LHGR limits and, therefore, the Specification is only required when the reactor is operating at  $\geq$  25% RTP.

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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 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.

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SURVEILLANCE        SR 3.2.3.1  
REQUIREMENTS

The LHGRs are required to be initially calculated within 12 hours after THERMAL POWER is  $\geq$  25% RTP and then every 24 hours thereafter. They are compared with the LHGR limits in the COLR to ensure that the reactor is operating within

(continued)

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.2.3.1 (continued)

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 under normal conditions. The 12 hour allowance after THERMAL POWER  $\geq$  25% RTP is achieved is acceptable given the inherent margin to operating limits at lower power levels.

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REFERENCES

1. UFSAR, Chapter 4.
  2. UFSAR, Chapter 15.
  3. NUREG-0800, Section 4.2.II A.2(g), Revision 2, July 1981.
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3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

LCD 3.2.1

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

A.2

APPLICABILITY: OPERATIONAL CONDITION 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, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than

L.A.1

ACTION B

25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits specified in the CORE OPERATING LIMITS REPORT.

a. At least once per 24 hours.

L.1

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

225%

L.1

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

L.2

3/4.2 POWER DISTRIBUTION LIMITS A.1

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

LCD 3.2.1

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: OPERATIONAL CONDITION 1 when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER. A.2

ACTION:

ACTION A — { With an APLHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours. L.1

ACTION B — {

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits specified in the CORE OPERATING LIMITS REPORT.

SR 3.2.1.1

- a. At least once per 24 hours. L.1
- b. Within 12 hours after completion of a THERMAL POWER ≥ 25% increase of at least 15% of RATED THERMAL POWER, and L.1
- c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR. L.2

DISCUSSION OF CHANGES  
ITS: 3.2.1 - AVERAGE PLANAR LINEAR HEAT GENERATION RATE

ADMINISTRATIVE

- A.1 In the conversion of the LaSalle 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-1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).
- A.2 The Applicability for CTS 3/4.2.1 is "OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER." With THERMAL POWER  $\geq$  25% RTP, the unit will always be in MODE 1. Therefore, it is unnecessary to state "OPERATIONAL CONDITION 1" in the Applicability of CTS 3/4.2.1 (ITS 3.2.1).

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA.1 The requirement in the CTS 3.2.1 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 3.2.1 ACTION A two 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.

"Specific"

- L.1 CTS 4.2.1.b 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

DISCUSSION OF CHANGES  
ITS: 3.2.1 - AVERAGE PLANAR LINEAR HEAT GENERATION RATE

TECHNICAL CHANGES - LESS RESTRICTIVE

- L.1 (cont'd) within 12 hours of reaching or exceeding 25% RTP will generally require that the Surveillance be performed sooner than the CTS 4.2.1.b 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 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.
- L.2 CTS 4.2.1.c, 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, 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

POWER DISTRIBUTION LIMITS

3/4.2.3 MINIMUM CRITICAL POWER RATIO

A.1

LIMITING CONDITION FOR OPERATION

LCD 3.2.2

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR limit specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY:

OPERATIONAL CONDITION 1 when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

A2

ACTION

a. With MCPR less than the applicable MCPR limit as determined for one of the conditions specified in the CORE OPERATING LIMITS REPORT.

ACTION A

1. Initiate corrective action within 15 minutes, and

LA1

2. Restore MCPR to within the required limit within 2 hours.

ACTION B

3. Otherwise, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

ACTION A

b. When operating in a condition not specified in the CORE OPERATING LIMITS REPORT, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within 4 hours.

L1

ACTION B

3/4.2.3 MINIMUM CRITICAL POWER RATIO

A.1

SURVEILLANCE REQUIREMENTS

SR3.2.2.1

4.2.3.1 MCPR shall be determined to be equal to or greater than the applicable MCPR limit specified in the CORE OPERATING LIMITS REPORT.

- a. At least once per 24 hours,
- b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

≥25%

L2

- c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

L3

SR3.2.2.2

4.2.3.2 The applicable MCPR limit shall be determined from the COLR based on:

- a. Technical Specification Scram Speed (TSSS) MCPR limits, or
- b. Nominal Scram Speed (NSS) MCPR limits if scram insertion times determined per surveillance 4.1.3.2 meet the NSS insertion times identified in the COLR.

LA2

Within 72 hours of completion of each set of scram testing, the results will be compared against the nominal scram speed (NSS) insertion times specified in the COLR, to verify the applicability of the transient analyses. Prior to initial scram time testing for an operating cycle, the MCPR operating limits used shall be based on the Technical Specification Scram Speeds (TSSS).

LA3

POWER DISTRIBUTION LIMITS

A.1

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

LC03.2.2

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR limit specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY:

OPERATIONAL CONDITION 1 when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

A.2

ACTION

**ACTION A** a. With MCPR less than the applicable MCPR limit as determined for one of the conditions specified in the CORE OPERATING LIMITS REPORT:

1. Initiate corrective action within 15 minutes, and

2. ~~Restore MCPR to within the required limit within 2 hours.~~

LA.1

**ACTION B** 3. Otherwise, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

**ACTION A** b. <sup>restore within 2 hours</sup> When operating in a condition not specified in the CORE OPERATING LIMITS REPORT, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within 4 hours.

L.1

POWER DISTRIBUTION LIMITS

A.1

3/4.2.3 MINIMUM CRITICAL POWER RATIO

SURVEILLANCE REQUIREMENTS

SR3.2.2.1

4.2.3.1 MCPR shall be determined to be equal to or greater than the applicable MCPR limit specified in the CORE OPERATING LIMITS REPORT.

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER <sup>≥25%</sup> increase of at least 15% of RATED THERMAL POWER, and L2

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR. L3

SR3.2.2.2

4.2.3.2 The applicable MCPR limit shall be determined from the COLR based on:

a. Technical Specification Scram Speed (TSSS) MCPR limits, or

b. Nominal Scram Speed (NSS) MCPR limits if scram insertion times determined per surveillance 4.1.3.2 meet the NSS insertion times identified in the COLR. LA2

Within 72 hours of completion of each set of scram testing, the results will be compared against the nominal scram speed (NSS) insertion times specified in the COLR to verify the applicability of the transient analyses. Prior to initial scram time testing for an operating cycle, the MCPR operating limits used shall be based on the Technical Specification Scram Speeds (TSSS). LA3

DISCUSSION OF CHANGES  
ITS: 3.2.2 - MINIMUM CRITICAL POWER RATIO

ADMINISTRATIVE

- A.1 In the conversion of the LaSalle 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-1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).
- A.2 The Applicability for CTS 3/4.2.3 is "OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER." With THERMAL POWER  $\geq$  25% RTP, the unit will always be in MODE 1. Therefore, it is unnecessary to state "OPERATIONAL CONDITION 1" in the Applicability of CTS 3/4.2.3 (ITS 3.2.2).

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA.1 The requirement in CTS 3.2.3, ACTION c, 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 (see Discussion of Changes for L.1) 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 of the method for comparing scram testing results and determining the MCPR limit are proposed to be relocated to the Bases of proposed SR 3.2.2.2 in the form of a discussion. The purpose of the comparison and determination is to appropriately apply the scram time data obtained after performing CTS 4.1.3.2.b (proposed SRs 3.1.4.1, 3.1.4.2, and 3.1.4.4 - see Discussion of Changes for LCO 3.1.4, "Control Rod Scram Times"). The proposed LCO 3.2.2 clearly

DISCUSSION OF CHANGES  
ITS: 3.2.2 - MINIMUM CRITICAL POWER RATIO

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.2 (cont'd) states that the requirement that all MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR. The proposed SR 3.2.2.1 specifies when the verification is required and SR 3.2.2.2 specifies when the MCPR limits must be recalculated based on actual scram speeds. Since all the requirements of the current LCO are continued to be maintained in the proposed specification, the proposed changes are considered adequate. 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 Bases Control Program described in Chapter 5 of the ITS.

LA.3 The CTS 4.2.3.2 requirement that, prior to initial scram time testing for an operating cycle, the MCPR operating limit be based on the Technical Specification Scram Speed (TSSS) is proposed to be relocated to the COLR. The limits for TSSS and Nominal Scram Speed (NSS) are presently located in the COLR. Therefore, it is reasonable and adequate that the required limit for MCPR for an operating cycle prior to initial scram time testing also be explicitly specified in the COLR with the other MCPR limits. Since all the requirements of the current LCO are continued to be maintained, the proposed change is considered adequate. 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 COLR will be controlled by the provisions of the COLR change control process described in Chapter 5 of the ITS.

"Specific"

L.1 The CTS 3.2.3 ACTION b Completion Time is proposed to be extended 2 hours to allow restoration of MCPR to within required limits. Currently, no time is provided to restore the MCPR limit prior to requiring a power reduction. The proposed 2 hour Completion Time for restoration allows appropriate actions to be evaluated by the operator and completed in a timely manner to restore the MCPR within limits. The condition not specified in the COLR may be evaluated to be bounded by a condition specified in the COLR which is permitted to be restored. This time is consistent for other situations wherein MCPR is less than the applicable MCPR limit as determined for one of the applicable conditions specified in the COLR and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the MCPR out of limits.

L.2 CTS 4.2.3.1.b 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

DISCUSSION OF CHANGES  
ITS: 3.2.2 - MINIMUM CRITICAL POWER RATIO

TECHNICAL CHANGES - LESS RESTRICTIVE

- L.2 (cont'd) the Surveillance be performed sooner than the CTS 4.2.3.1.b 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 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.
- L.3 CTS 4.2.3.1.c, 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

POWER DISTRIBUTION LIMITS

3/4.2.4 LINEAR HEAT GENERATION RATE [A.1]

LIMITING CONDITION FOR OPERATION

CCO3.2.3 3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: ~~OPERATIONAL CONDITION 1~~ when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER. [A.2]

ACTION:

ACTION A

ACTION B

With the LHGR of any fuel rod exceeding the limit, ~~initiate corrective action~~ within 15 minutes and restore the LHGR to within the limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours. [LA.1]

SURVEILLANCE REQUIREMENTS

SR3.2.3.1 4.2.4 LHGR's shall be determined to be equal to or less than the limit:

a. At least once per 24 hours,

b. Within 12 hours after ~~completion of a~~ THERMAL POWER <sup>225%</sup> increase of ~~at least 15%~~ of RATED THERMAL POWER, and [L.1]

c. Initially and at least once per 12 hours when the reactor is operating on a LIMITING CONTROL ROD PATTERN for LHGR. [L.2]

POWER DISTRIBUTION LIMITS

3/4.2.4 LINEAR HEAT GENERATION RATE

A.1

LIMITING CONDITION FOR OPERATION

LC03.2.3

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: ~~OPERATIONAL CONDITION 1~~ when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

A.2

ACTION:

ACTION A

ACTION B

With the LHGR of any fuel rod exceeding the limit, ~~initiate corrective action within 15 minutes~~ and restore the LHGR to within the limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

LA.1

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

4.2.4 LHGR's shall be determined to be equal to or less than the limit:

- a. At least once per 24 hours,
- b. Within 12 hours after ~~completion of a~~ THERMAL POWER <sup>>25%</sup> ~~(increase of)~~ ~~at least 15%~~ of RATED THERMAL POWER, and
- c. Initially and at least once per 12 hours when the reactor is operating on a LIMITING CONTROL ROD PATTERN for LHGR.

L.1

L.2

DISCUSSION OF CHANGES  
ITS: 3.2.3 - LINEAR HEAT GENERATION RATE

ADMINISTRATIVE

- A.1 In the conversion of the LaSalle 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-1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).
- A.2 The Applicability for CTS 3/4.2.4 is "OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER." With THERMAL POWER  $\geq$  25% RTP, the unit will always be in MODE 1. Therefore, it is unnecessary to state "OPERATIONAL CONDITION 1" in the Applicability of CTS 3/4.2.4 (ITS 3.2.3).

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA.1 The requirement in the CTS 3.2.4 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 3.2.3 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.

"Specific"

- L.1 CTS 4.2.4.b 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

DISCUSSION OF CHANGES  
ITS: 3.2.3 - LINEAR HEAT GENERATION RATE

TECHNICAL CHANGES - LESS RESTRICTIVE

- L.1 (cont'd) parameter within 12 hours of reaching or exceeding 25% RTP will generally require that the Surveillance be performed sooner than the CTS 4.2.4.b 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 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.
- L.2 CTS 4.2.4.c, 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

DISCUSSION OF CHANGES  
ITS: SECTION 3.2 - POWER DISTRIBUTION LIMITS BASES

The Bases of the current Technical Specifications for this section (pages B 3/4 2-1 through B 3/4 2-6) have been completely replaced by revised Bases that reflect the format and applicable content of the LaSalle 1 and 2 ITS Section 3.2, consistent with the BWR Standard Technical Specifications, NUREG-1434, Rev. 1. The revised Bases are as shown in the LaSalle 1 and 2 ITS Bases. In addition, page B 3/4 2-2, which is a blank page, has been removed.

<CTS>

3.2 POWER DISTRIBUTION LIMITS

3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

<LCO 3.2.1>

LCO 3.2.1 All APLHGRs shall be less than or equal to the limits specified in the COLR.

<Appl 3.2.1>

APPLICABILITY: THERMAL POWER  $\geq$  25% RTP.

**ACTIONS**

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.2.1 Act> A. Any APLHGR not within limits.	A.1 Restore APLHGR(s) to within limits.	2 hours
<3.2.1 Act> B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

**SURVEILLANCE REQUIREMENTS**

SURVEILLANCE	FREQUENCY
<4.2.1> <4.2.1.a> <4.2.1.b> 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

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1  
ITS: 3.2.1 - AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1. There are no deviations from NUREG-1434, Revision 1, for proposed Specification 3.2.1.

<CTS>

3.2 POWER DISTRIBUTION LIMITS

3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

<LCO 3.2.3> LCO 3.2.2 All MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR.

<Appl 3.2.3> APPLICABILITY: THERMAL POWER  $\geq$  25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>&lt;3.2.3 Act a&gt; &lt;3.2.3 Act a.2&gt; &lt;3.2.3 Act b&gt;</p> <p>A. Any MCPR not within limits.</p>	<p>A.1 Restore MCPR(s) to within limits.</p>	2 hours
<p>&lt;3.2.3 Act a.3&gt; &lt;3.2.3 Act b&gt;</p> <p>B. Required Action and associated Completion Time not met.</p>	<p>B.1 Reduce THERMAL POWER to &lt; 25% RTP.</p>	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>&lt;4.2.3.1.a&gt; &lt;4.2.3.1.b&gt;</p> <p>SR 3.2.2.1 Verify all MCPRs are greater than or equal to the limits specified in the COLR.</p>	<p>Once within 12 hours after <math>\geq</math> 25% RTP</p> <p><u>AND</u></p> <p>24 hours thereafter</p>

→  
Insert SR 3.2.2.2 [1]

<CTS>

Insert SR 3.2.2.2

SR 3.2.2.2

Determine the MCPR limits.

Once within 72 hours after  
each completion of SR 3.1.4.1

<4.2,3.2>

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

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1  
ITS: 3.2.2 - MINIMUM CRITICAL POWER RATIO

1. LaSalle 1 and 2 currently have a requirement to calculate the MCPR limit which is similar to the requirements of BWR/4 type plants and reflected in NUREG-1433, Revision 1, SR 3.2.2.2, as modified by TSTF-229, Rev. 0. The requirement to perform the surveillance once within 72 hours after each completion of SR 3.1.4.1, SR 3.1.4.2, and SR 3.1.4.4 has been added to be consistent with the current licensing basis of LaSalle 1 and 2.

<CTS>

3.2 POWER DISTRIBUTION LIMITS

3.2.3 LINEAR HEAT GENERATION RATE (LHGR) (Optional) 1

<LCO 3.2.4>

LCO 3.2.3 All LHGRs shall be less than or equal to the limits specified in the COLR.

<Appl 3.2.4>

APPLICABILITY: THERMAL POWER  $\geq$  25% RTP.

ACTIONS

<3.2.4 Act>

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

<3.2.4 Act>

SURVEILLANCE REQUIREMENTS

<4.2.4.4.2.4.a.4.2.4.b>

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  <u>AND</u> 24 hours thereafter

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1  
ITS: 3.2.3 - LINEAR HEAT GENERATION RATE

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.

3.2 POWER DISTRIBUTION LIMITS

3.2.4 Average Power Range Monitor (APRM) Gain and Setpoints (Optional)

- LCO 3.2.4
- a. MFLPD shall be less than or equal to Fraction of RTP; or
  - 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  $\geq 100\%$  times MFLPD.

APPLICABILITY: THERMAL POWER  $\geq 25\%$  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
<p>SR 3.2.4.1</p> <p>-----NOTE----- Not required to be met if SR 3.2.4.2 is satisfied for LCO 3.2.4 Item b or c requirements.</p> <p>-----</p> <p>Verify MFLPD is within limits.</p>	<p>Once within 12 hours after <math>\geq 25\%</math> RTP</p> <p><b>AND</b></p> <p>24 hours thereafter</p>
<p>SR 3.2.4.2</p> <p>-----NOTE----- Not required to be met if SR 3.2.4.1 is satisfied for LCO 3.2.4 Item a requirements.</p> <p>-----</p> <p>Verify APRM setpoints or gains are adjusted for the calculated MFLPD.</p>	<p>12 hours</p>

1

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1  
ISTS: 3.2.4 - APRM GAIN AND SETPOINTS

1. This optional Specification is not used because LaSalle 1 and 2 utilize ARTS based APRM limits.

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

criteria

limits specified in 10 CFR 50.46.

are met

<Insert B3.2.1-BK6D>

1

APPLICABLE SAFETY ANALYSES

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

u

1 and 2

1

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 LHGR 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. 2 and 3). Flow dependent APLHGR limits are determined using the three dimensional BWR simulator code (Ref. 4) to analyze slow flow runout transients. The flow dependent multiplier, MAPFAC, is dependent on the maximum core flow runout capability. MAPFAC curves are provided based on the maximum credible flow runout transient for Loop Manual and Non Loop Manual operation. The result of a single failure or single operator error during Loop Manual operation is the runout of only one loop because both recirculation loops are under independent control. Non Loop Manual operational modes allow simultaneous runout of both loops because a single controller regulates core flow.

1

(continued)

Insert B 3.2.1-BKGD

Additionally, for GE fuel types, APLHGR limits are specified to ensure that the fuel design limits identified in Reference 1 are not exceeded during anticipated operational occurrences (A00s).

BASES

APPLICABLE  
SAFETY ANALYSES  
(continued)

Based on analyses of limiting plant transients (other than core flow increases) over a range of power and flow conditions, power dependent multipliers,  $MAPFAC_p$ , are also generated. Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which turbine stop valve closure and turbine control valve fast closure scram signals 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 AOs. A complete discussion of the analysis code is provided in References 1 and 3.

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,

LOCA analyses are ~~then~~ performed to ensure that the ~~above~~ ~~determined~~ APLHGR limits are adequate to meet the (PCT) and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using calculational models that are consistent with the requirements of 10 CFR 50, Appendix K. A complete discussion of the analysis code is provided in Reference 3. 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. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by ~~(25)~~ local peaking factor. A

peak cladding temperature

conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.

the minimum anticipated

A conservative multiplier is applied to the exposure dependent APLHGR limits for two loop operation

For single recirculation loop operation, ~~the MAPFAC multiplier is limited to a maximum of 0.86 (Ref. 2)~~. This ~~limit~~ is due to the conservative analysis assumption of an earlier departure from nucleate boiling with one recirculation loop available, resulting in a more severe cladding heatup during a LOCA.

additional limitation

Insert B3.2.1 ASA

The APLHGR satisfies Criterion 2 of ~~the NRC Policy Statement~~.

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

LCO

The APLHGR limits specified in the COLR are the result of fuel design, DBA, and transient analyses. For two

(continued)

For Siemens Power Corporation fuel, APLHGR limits are typically set high enough such that the LHGR limits are more limiting than the APLHGR limits.

Insert B 3.2.1 ASA

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 the UFSAR, Chapters 4, 6, and 15, and in Reference 1. 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 LHGR 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.

BASES

dependent on exposure [1]

LCO  
(continued)

recirculation loops operating, the limit is determined by multiplying the smaller of the MAPFAC<sub>A</sub> and MAPFAC<sub>B</sub> factors times the exposure dependent APLHGR limits. With only one recirculation loop in operation, in conformance with the requirements of LCO 3.4.1, "Recirculation Loops Operating," the limit is determined by multiplying the exposure dependent APLHGR limit by the smallest of MAPFAC<sub>A</sub>, MAPFAC<sub>B</sub>, and 0.86, where 0.86 has been determined by a specific single recirculation loop analysis (Ref. 2). [1]

[1]  
a conservative multiplier

APPLICABILITY

The APLHGR limits are primarily derived from fuel design evaluations and LOCA and transient analyses that are assumed to occur at high power levels. Design calculations (Ref. 4) 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 (IRM) 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 operates with (substantial) margin to the APLHGR limits; thus, this LCO is not required. [1]

and average power range monitor (APRM) scram function [1]

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 is taken to restore the APLHGR(s) to within the required limits such that the plant will be operating within analyzed conditions and within the 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

(continued)

BASES

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ACTIONS

B.1 (continued)

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.

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SURVEILLANCE  
REQUIREMENTS

SR 3.2.1.1

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

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REFERENCES

1. [Plant specific current cycle safety analysis].
2. FSAR, [Chapter 15, Appendix C].
3. FSAR, [Chapter 15, Appendix D].
4. XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors, Neutronics Methods for Design and Analysis," Volume 1, June 1981.
5. XN-NF-80-19(A), "Exxon Nuclear Methodology for Boiling Water Reactors, ECES Evaluation Model," Volume 2, Revision 1, June 1981.

<Insert B3.2.1 Ref>

Insert B 3.2.1 Ref

1. NEDO-24011-P-A, "General Electric Standard Application for Reactor Fuel," (as specified in Technical Specification 5.6.5).
2. EMF-94-217(NP), Revision 1, "Boiling Water Reactor Licensing Methodology Summary," November 1995.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, 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, or analysis description, or licensing basis description.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

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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, 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 experiences boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

1 are expected to

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 ~~the PSAR Chapters 4, 6, and 15, and~~ 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 ( $\Delta$ CPR). When the largest  $\Delta$ CPR is added to the MCPR SL, the required operating limit MCPR is obtained.

(continued)

BASES

1 With key physics response inputs benchmarked

APPLICABLE SAFETY ANALYSES (continued)

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state (MCPR<sub>f</sub> and MCPR<sub>p</sub>, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency (Refs. 3, 4, and 5). Flow dependent MCPR limits are determined by steady state thermal hydraulic methods using the three dimensional BWR simulator code (Ref. 8) and the multichannel thermal hydraulic code (Ref. 9). MCPR<sub>f</sub> curves are provided based on the maximum credible flow runout transient for Loop Manual and Non Loop Manual operation. The result of a single failure or single operator error during Loop Manual operation is the runout of only one loop because both recirculation loops are under independent control. Non Loop Manual operational modes allow simultaneous runout of both loops because a single controller regulates core flow.

9 1

1 as identified in the UFSAR, Chapter 15 (Ref. 5)

8 MCPR<sub>f</sub> curves are provided based on the maximum credible flow runout transient for Loop Manual and Non Loop Manual operation. The result of a single failure or single operator error during Loop Manual operation is the runout of only one loop because both recirculation loops are under independent control. Non Loop Manual operational modes allow simultaneous runout of both loops because a single controller regulates core flow.

9

1 to analyze slow flow runout transients on a cycle-specific basis

MCPR<sub>f</sub> curves are provided based on the maximum credible flow runout transient for Loop Manual and Non Loop Manual operation. The result of a single failure or single operator error during Loop Manual operation is the runout of only one loop because both recirculation loops are under independent control. Non Loop Manual operational modes allow simultaneous runout of both loops because a single controller regulates core flow.

1 Insert ASA

Power dependent MCPR limits (MCPR<sub>p</sub>) are determined by the three dimensional BWR simulator code and the one dimensional transient code (Ref. 8). 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 scram trips are bypassed, high and low flow MCPR<sub>p</sub> operating limits are provided for operating between 25% RTP and the previously mentioned bypass power level.

1 on a cycle-specific basis. These limits are established to protect the core from plant transients other than core flow increases, including pressurization and local control rod withdrawal events.

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 MCPR operating limits are determined by the larger of the MCPR<sub>f</sub> and MCPR<sub>p</sub> limits.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a slow 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.

(continued)

1 MCPR operating limits which include the effects of analyzed equipment out-of-service are also included in the COLR.

Insert ASA

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 power/flow state is the greater of the rated conditions MCPR operating limit or the power 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)

Statistical analyses documented in Reference 9 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 (Ref. 5) 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 (IRM) 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.

1

1  
and average power range monitor (APRM)

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

1

← Insert B3.2.2 SR 1

REFERENCES

1. NUREG-0562, June 1979.

← Insert B 3.2.2 Ref 1

2. [Plant specific current cycle safety analysis].
3. FSAR, [Appendix 15B].
4. FSAR, [Appendix 15C].
5. FSAR, [Appendix 15D].
6. XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors, Neutronics Methods for Design and Analysis," Volume 1 (as supplemented).
7. XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX Thermal Limits Methodology Summary Description," Volume 3, Revision 2, January 1987.
8. XN-NF-79-71(P), "Exxon Nuclear Plant Methodology for Boiling Water Reactors," Revision 2, November 1981.
9. "BWR/6 Generic Rod Withdrawal Error Analysis," General Electric Standard Safety Analysis Report, GESSAR-II, Appendix 15B.

1

### Insert B 3.2.2 SR

#### SR 3.2.2.2

Because the transient analyses may take credit for conservatism in the control rod 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 either on the applicable limit associated with scram times of LCO 3.1.4, "Control Rod Scram Times," or the realistic scram times. The scram time dependent MCPR limits are contained in the COLR. This determination must be performed within 72 hours after each set of control rod 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 control rod scram speed distribution expected during the fuel cycle.

### Insert B 3.2.2 Ref

2. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (as specified in Technical Specification 5.6.5).
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-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-1434, 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, or analysis description.

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 the 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 anticipated operating conditions identified in Reference 1.

2 normal operations and

2 S 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~~ mechanisms that could cause fuel damage during operational transients and that are considered in fuel evaluations are:

2 2 A normal operations and

- 2 a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO<sub>2</sub> pellets; and 2
- b. Severe overheating of the fuel rod cladding caused by inadequate cooling. 2
- 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).
- 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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

the operating limit specified in the COLR. The analysis also includes allowances for short term transient operation above the operating limit to account for AOs, plus an excursions allowance for densification power spiking. While still remaining within the AOs limits  
The LHGR satisfies Criterion 2 of the NRC Policy Statement.  
10 CFR 50.36 (c)(2)(i)

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 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 must be reduced to < 25% RTP within 4 hours. The allowed

(continued)

BASES

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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.

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SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

4 The LHGR <sup>5</sup> ~~is~~ <sup>are</sup> required to be initially calculated within 12 hours after THERMAL POWER is  $\geq$  25% RTP and then every 24 hours thereafter. ~~It is~~ compared with the ~~specified~~ <sup>LHGR</sup> limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. <sup>They are</sup> The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution under normal conditions. The 12 hour allowance after THERMAL POWER  $\geq$  25% RTP is achieved is acceptable given the <sup>2</sup> ~~large~~ inherent margin to operating limits at lower power levels.

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REFERENCES

1. ~~[Non GE Fuel Analysis]~~ 2
  2. FSAR, Chapter ~~4~~ 3
  3. NUREG-0800, Section II A.2(g), Revision 2, July 1981. 4.2. 2
  2. UFSAR, Chapter 15. 2
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JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, 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, or analysis 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.

**B 3.2 POWER DISTRIBUTION LIMITS**

**B 3.2.4 Average Power Range Monitor (APRM) Gain and Setpoints**

**BASES**

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**BACKGROUND**

The OPERABILITY of the APRMs and their setpoints is an initial condition of all safety analyses that assure rod insertion upon reactor scram. Applicable GDCs are GDC 10, "Reactor Design"; GDC 13, "Instrumentation and Control"; GDC 20, "Protection System Functions"; and GDC 29, "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.

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 MFPLD to the Fraction of RTP (FRTP) where FRTP is the measured THERMAL POWER divided by the RTP. Excessive power peaking exists when:

$$\frac{\text{MFPLD}}{\text{FRTP}} > 1,$$

indicating that MFPLD 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 gain adjustment on the APRMs or adjustment of the APRM setpoints. Either of these adjustments has effectively the same result as maintaining MFPLD less than or equal to FRTP and thus maintains RTP margins for APLHGR and MCPR.

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,

(continued)

**BASES**

**BACKGROUND  
(continued)**

the margin to scram or rod blocks 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.

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**

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.

FSAR safety analyses (Refs. 2 and 3) concentrate on the rated power condition for which the minimum expected margin

(continued)



**BASES**

**APPLICABLE  
SAFETY ANALYSES  
(continued)**

to the operating limits (APLHGR and MCPR) occurs. 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 pretransient 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.

The APRM gain and setpoints satisfy Criteria 2 and 3 of the NRC Policy Statement.

**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

(continued)

1

**BASES**

**LCO**  
(continued)

c. Increasing the APRM gains to cause the APRM to read greater than 100(%) times 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.

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 the APRM gain or setpoints is equivalent to maintaining MFLPD less than or equal to F RTP, as stated in the LCO.

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.

**APPLICABILITY**

The MFLPD limit, APRM gain adjustment, or 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 plant is operating at  $\geq$  25% RTP.

**ACTIONS**

**A.1**

If the APRM gain or setpoints are not within limits while the MFLPD has exceeded F RTP, the margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit

(continued)

**BASES**

**ACTIONS**

**A.1 (continued)**

may be reduced. Therefore, prompt action should be taken to restore the MFLPD 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 to within limits or the APRM gain or setpoints 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 the APRM gain or setpoints cannot be restored to within their 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 3.2.4.1 and SR 3.2.4.2**

The MFLPD is required to be calculated and compared to F RTP or APRM gain or setpoints to ensure that the reactor is operating within the assumptions of the safety analysis. These SRs are required only to determine the MFLPD and, assuming MFLPD is greater than F RTP, the appropriate gain or setpoint, and is not intended to be a CHANNEL FUNCTIONAL TEST for the APRM gain or flow biased neutron flux scram circuitry. 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). The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER  $\geq$  25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

(continued)

		APRM Gain and Setpoints B 3.2.4
<b>BASES</b>		
<b>SURVEILLANCE REQUIREMENTS</b>	<b>SR 3.2.4.1 and SR 3.2.4.2 (continued)</b> 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.	
<b>REFERENCES</b>	1. 10 CFR 50, Appendix A, GDC 10, GDC 13, GDC 20, and GDC 29. 2. FSAR, Section [ ]. 3. FSAR, Section [ ].	

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JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1  
ISTS BASES: 3.2.4 - APRM GAIN AND SETPOINTS

1. This Bases section has been deleted since the associated optional Specification is not used.

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.

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: SECTION 3.2 - POWER DISTRIBUTION LIMITS

**"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 NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: SECTION 3.2 - POWER DISTRIBUTION LIMITS

"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 Standard Technical Specifications, NUREG-1434, 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 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.

NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: 3.2.1 - AVERAGE PLANAR LINEAR HEAT GENERATION RATE

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.

NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: 3.2.2 - MINIMUM CRITICAL POWER RATIO

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 in the completion time to allow 2 hours to restore MCPR(s) provides reasonable time so that the operator can evaluate the condition and take the appropriate actions. 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 the completion time will not significantly increase the probability of an accident previously identified. While MCPR being within limits is assumed to mitigate the consequences of an event, the consequences of an event during the additional 2 hours is the same as the consequences of an event during the current time provided to reduce power. Therefore, 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?

The increased completion time to restore MCPR(s) within limits is acceptable based on the small probability of an event occurring during this time period. At the same time the Fuel Integrity Safety Limit (CTS 2.1.2 and proposed Specification 2.1.1.2) ensures that no significant fuel cladding damage will occur during steady condition since the MCPR must be maintained above the Safety Limit. The 2 hour Completion Time provides adequate time to evaluate and to take the corrective action to restore the MCPR(s) above the LCO limits. Due to the small probability of an event occurring during this time period and since the Fuel Integrity Safety Limit must be met during this time period, this change does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: 3.2.2 - MINIMUM CRITICAL POWER RATIO

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 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 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.

NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: 3.2.2 - MINIMUM CRITICAL POWER RATIO

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 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 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.

NO SIGNIFICANT HAZARDS CONSIDERATION  
ITS: 3.2.3 - LINEAR HEAT GENERATION RATE

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.

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.