ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
			COND
~~2~~ 1. PRIMARY CONTAINMENT ISOLATION			
a. Reactor Vessel Water Level - Low, Level 2 (Division 1 & 2)	2	1, 2, 3 and ~~#~~ (b)	~~20~~ H/L
e ~~b~~. Reactor Vessel Water Level - Low, Level 2 (Division 3)	4[d]	1, 2, 3 and ~~#~~ (b)	28 F/L
b ~~c~~. Drywell Pressure - High (Division 1 & 2)	2	1, 2, 3	20 H F
f ~~d~~. Drywell Pressure - High (Division 3)	4[d]	1, 2, 3	28 F
g ~~e~~. Containment and Drywell Purge Exhaust Plenum Radiation - High	2[b]	1, 2, 3 and ~~*~~ (c)	~~21~~ F/K
c ~~f~~. Reactor Vessel Water Level - Low, Level 1	2	1, 2, 3 and ~~#~~ (b)	~~20~~ F/L
h ~~g~~. Manual Initiation (Division 1 & 2)	2[c]	1, 2, 3 and ~~*~~ (c)	~~22~~ G/K
h. Manual Initiation (Division 3)	1[e]	1, 2, 3 and *	28
1 ~~2~~. MAIN STEAM ISOLATION			
a. Reactor Vessel Water Level - Low, Level 1	2	1, 2, 3	~~20~~ D
b. Main Steam Line Radiation - High	2	1, 2	23
b ~~c~~. Main Steam Line Pressure - Low	2	1	~~24~~ E
c ~~d~~. Main Steam Line Flow - High	2/line	1, 2, 3	~~23~~ D
d ~~e~~. Condenser Vacuum - Low	2	1, 2**, 3**	~~23~~ D
e ~~f~~. Main Steam Line Tunnel Temperature - High	2	1, 2, 3	~~23~~ D
~~g~~. Main Steam Line Tunnel Δ Temperature - High	2	1, 2, 3	~~23~~
f. ~~g~~ ~~h~~. Turbine Building Main Steam Line Temperature - High	2	1, 2, 3	~~23~~ D
g. ~~K~~ ~~i~~. Manual Initiation	2	1, 2, 3	~~22~~ G/H

Table 3.3.6.1-1 Item 3.h

LA5

A9

(a)

A2

3.3.6.1 C:12

R1

#
—

L4

PERRY - UNIT 1

3/4 3-11

Amendment No. ~~44~~,50

9505250171 950518
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P PDR

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Attachment 2
Page 1

LCO 3.3.6.1

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

L17

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
3. SECONDARY CONTAINMENT ISOLATION			
a. Reactor Vessel Water Level - Low, Level 2	2	1, 2, 3 and #	25
b. Drywell Pressure - High	2	1, 2, 3	25
c. Manual Initiation	2	1, 2, 3	22
	2	*	25
4. REACTOR WATER CLEANUP SYSTEM ISOLATION			COND
a. Δ Flow - High	1	1, 2, 3	F/H 27
b. Δ Flow Timer	1	1, 2, 3	27
c, d, e, f, g, h ~~c.~~ Equipment Area Temperature - High (A13)	1	1, 2, 3	27
d. Equipment Area Δ Temperature - High	1	1, 2, 3	27
j g ~~e.~~ Reactor Vessel Water Level - Low, Level 2 (3.3.6.1 C:12 — R1)	2 (#)	1, 2, 3	27
i ø ~~f.~~ Main Steam Line Tunnel Ambient Temperature - High	1	1, 2, 3	27
g. Main Steam Line Tunnel Δ Temperature - High	1	L15 1, 2, 3	
k ~~h.~~ SLCS Initiation	1	1, 2, 3	I 27
l ~~i.~~ Manual Initiation	2	1, 2, 3	G/H 26

A14

q

LCO 3.3.6.1

~~TABLE 3.3.2-1 (Continued)~~ 2

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
3 ~~B.~~ REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION			
a. RCIC Steam Line Flow - High	1	1, 2, 3	F/H 27
c ~~b.~~ RCIC Steam Supply Pressure - Low	1	1, 2, 3	27
d ~~c.~~ RCIC Turbine Exhaust Diaphragm Pressure - High	2	1, 2, 3	27
e ~~d.~~ RCIC Equipment Room Ambient Temperature - High	1	1, 2, 3	27
~~e. RCIC Equipment Room Δ Temperature - High~~ A8	~~1~~	~~1, 2, 3~~	~~27~~ 2
f. Main Steam Line Tunnel Ambient Temperature - High	1	1, 2, 3	27
g. Main Steam Line Tunnel Δ Temperature - High	1	1, 2, 3	27
g ~~h.~~ Main Steam Line Tunnel Temperature Timer	1	1, 2, 3	27
h ~~i.~~ RHR Equipment Room Ambient Temperature - High	1/Area	1, 2, 3	27
j. RHR Equipment Room Δ Temperature - High	1/Area	1, 2, 3	27
b ~~k.~~ RCIC Steam Line Flow High Timer	1	1, 2, 3	27
j ~~l.~~ Drywell Pressure - High	1	1, 2, 3	27
k ~~m.~~ Manual Initiation	1	1, 2, 3	G/H 26 L8

3.3.6.1 C:12 R1 #

LCO 3.3.6.1.

3 i. <RHR/RCIC Steam Line flow-High from 6.c (RHR System Isolation)>

~~TABLE 3.3.2-1 (Continued)~~

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
5 ~~5.~~ RHR SYSTEM ISOLATION			
a. RHR Equipment Area Ambient Temperature - High (3.3.6.1 C:12) R1 (F)	1/Area	L13 ~~1~~, 2, 3	~~28~~ F/H A4
~~b. RHR Equipment Area Δ Temperature - High~~	~~1/Area~~	~~1, 2, 3~~	~~28~~
TO RCIC ISOLATION 3i ~~c.~~ RHR/RCIC Steam Line Flow - High	1	1, 2, 3	F/H ~~28~~ (3.3.6.1 C:21, 22, 45)
b ~~d.~~ Reactor Vessel Water Level - Low, Level 3	2 (f)	L13 ~~1, 2, 3~~ M5 4(f), 5(f)	28 J - M4
c ~~e.~~ Reactor Vessel ~~(RHR Cut-in Permissive) Pressure - High~~ LA2	2	1, 2, 3	~~28~~ F/H A4
d ~~f.~~ Drywell Pressure - High	2	1, 2, 3	~~28~~ F/H
e ~~g.~~ Manual Initiation	2	1, 2, 3	~~26~~ G/H

LCO 3.3.6.1

LCO 3.3.6.1

Table 3.3.6.1-1
~~TABLE 3.3.2-3~~

LSR 3.3.6.1.6

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level - Low, Level 2	NA
b. Drywell Pressure - High	NA
c. Containment and Drywell Purge Exhaust Plenum Radiation - High(b)	$< 10^{(a)}$ L18 ~~LA1~~ 3.3.6.1 C:35
d. Reactor Vessel Water Level - Low, Level 1	NA
e. Manual Initiation	NA
2. MAIN STEAM LINE ISOLATION	LA1
a. Reactor Vessel Water Level - Low, Level 1	$\leq$ 1.0* / $\leq 10^{(a)}$**
LA5 A9 b. Main Steam Line Radiation - High(b)	$\leq$ 1.0* / $\leq 10^{(a)}$**
c. Main Steam Line Pressure - Low	$\leq$ 1.0* / $\leq 10^{(a)}$**
d. Main Steam Line Flow - High	$\leq$ 0.5* / $\leq 10^{(a)}$** L18
e. Condenser Vacuum - Low	NA
f. Main Steam Line Tunnel Temperature - High	NA
g. Main Steam Line Tunnel Δ Temperature - High	NA R1 3.3.6.1 C:12 #1
h. Turbine Building Main Steam Line Temperature - High	NA
i. Manual Initiation	NA
3. SECONDARY CONTAINMENT ISOLATION L17	
a. Reactor Vessel Water Level - Low, Level 2	NA
b. Drywell Pressure - High	NA
c. Manual Initiation	NA
4. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. Δ Flow - High	NA
b. Δ Flow Timer	NA
c. Equipment Area Temperature - High	NA
d. Equipment Area Δ Temperature - High	NA
e. ~~Reactor Vessel Water Level - Low, Level 2~~	NA
f. Main Steam Line Tunnel Ambient Temperature - High	NA
g. Main Steam Line Tunnel Δ Temperature - High	NA
h. SLCS Initiation	NA
i. Manual Initiation	NA

LCO 3.3.6.1

~~TABLE 3.3.2-3~~ (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	
a. RCIC Steam Line Flow - High	NA
b. RCIC Steam Supply Pressure - Low	NA
c. RCIC Turbine Exhaust Diaphragm Pressure - High	NA
d. RCIC Equipment Room Ambient Temperature - High	NA
~~e. RCIC Equipment Room Δ Temperature - High~~	~~NA~~ A8
f. Main Steam Line Tunnel Ambient Temperature - High	NA
g. Main Steam Line Tunnel Δ Temperature - High	NA R1 3.3.6.1 C.1.3 #1
h. Main Steam Line Tunnel Temperature Timer	NA
i. RHR Equipment Room Ambient Temperature - High	NA
j. RHR Equipment Room Δ Temperature - High	NA
k. RCIC Steam Line Flow High Timer	NA
l. Drywell Pressure - High	NA
m. Manual Initiation	NA
6. RHR SYSTEM ISOLATION	
a. RHR Equipment Area Ambient Temperature - High	NA
b. RHR Equipment Area Δ Temperature - High	NA
c. RHR/RCIC Steam Line Flow - High	NA
d. Reactor Vessel Water Level - Low, Level 3	NA
e. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA
f. Drywell Pressure - High	NA
g. Manual Initiation	NA

LA1 e

3.3.6.1 C.3.8 L18

(a) Isolation system instrumentation response time specified includes the diesel generator starting and sequence loading delays. e <SR 3.3.6.1.6 Note> e

(b) Radiation detectors are exempt from response time testing. Response time shall be measured from detector output or the input of the first electronic component in the channel. LA1

LA2
*Isolation system instrumentation response time for MSIVs only. No diesel generator delays assumed.

**Isolation system instrumentation response time for associated valves except MSIVs. e

LA1
#Isolation system instrumentation response time specified for the Trip Function actuating each containment isolation valve shall be added to the isolation time for each valve to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

SEE MARKUP OF TABLE 3.3.2.1-1

TABLE ~~4.3.2.1-1~~ 3.3.6.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK (SR 3.3.6.1.1)	CHANNEL FUNCTIONAL TEST (SR 3.3.6.1.2)	CHANNEL CALIBRATION (SR 3.3.6.1.3, SR 3.3.6.1.4, SR 3.3.6.1.5)	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
~~1.~~ 2. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low, Level 2	S -1	~~M~~ Q -2	4 R(b) -3	1, 2, 3 and #
b. Drywell Pressure - High	S -1	~~M~~ Q -2	4 R(b) -3	1, 2, 3
c. Containment and Drywell Purge Exhaust Plenum Radiation - High	S -1	~~M~~ Q -2	R -5	1, 2, 3 and *
d. Reactor Vessel Water Level - Low, Level 1	S -1	~~M~~ Q -2	4 R(b) -3	1, 2, 3 and #
e. Manual Initiation	NA	R -5	NA	1, 2, 3 and *
~~2.~~ 1. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 1	S -1	~~M~~ Q -2	4 R(b) -3	1, 2, 3
~~b. Main Steam Line Radiation - High~~	~~S~~	~~M~~	~~R~~	~~1, 2~~
~~c.~~ b. Main Steam Line Pressure - Low	S -1	~~M~~ Q -2	4 R(b) -3	1
~~d.~~ c. Main Steam Line Flow - High	S -1	~~M~~ Q -2	4 R(b) -3	1, 2, 3
~~e.~~ d. Condenser Vacuum - Low	S -1	~~M~~ Q -2	4 R(b) -3	1, 2**, 3**
~~f.~~ e. Main Steam Line Tunnel Temperature - High	S -1	~~M~~ Q -2	R -4	1, 2, 3
~~g.~~ f. Main Steam Line Tunnel Δ Temperature - High	S -1	~~M~~ Q -2	R -4	1, 2, 3
~~h.~~ g. Turbine Building Main Steam Line Temperature - High	S -1	~~M~~ Q -2	R -4	1, 2, 3
~~i.~~ h. Manual Initiation	NA	R -5	NA	1, 2, 3

LA5

A9

A7

A7

See markup of Table 3.3.2.1-1 for isolation ΔT changes

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Attachment 2
Page 7

DISCUSSION OF CHANGES
CTS: 3.3.2 - ISOLATION ACTUATION INSTRUMENTATION

ADMINISTRATIVE
(continued)

A.9 These changes to this section have been previously proposed in a letter to the NRC, PY-CEI/NRR-1439 L, dated September 28,1992, or have been previously approved by the NRC in Amendment 44 to the license. The letter proposed eliminating the Main Steam Line Radiation Monitors (MSLRM) instrumentation trip signal to the MSIV. However the instrumentation requirement was proposed to remain in the specifications to fulfill a note requirement to the Table. Amendment 44 to the license removed numerous notes to the Isolation instrumentation table, including the note affecting the MSLRMs. The No Significant Hazards Consideration for the proposed changes is still valid. Therefore this change is considered administrative for purposes of this submittal.

A.10 This comment number is not used for this station.

A.11 This comment number is not used for this station.

A.12 This comment number is not used for this station.

A.13 The Area Temperature and Area Differential Temperature Functions are separately identified in the proposed Specifications by area. Since this does not change the required instrumentation but is only a change in presentation of the same requirements, this change is considered administrative.

A.14 The current ACTION for this function requires both isolation and declaration of the affected system as inoperable. Since the instrumentation for isolation of RWCU is to assure OPERABILITY of the SLCS, the "affected system" can be either the SLCS or the RWCU. These choices are reflected in the proposed Required Actions. Therefore, this change is considered administrative.

RELOCATED SPECIFICATIONS

R.1 This comment number is not used for this station.

Insert 11a #1

#1

DISCUSSION OF CHANGES
CTS: 3.3.2 - ISOLATION ACTUATION INSTRUMENTATION

(Insert 11a)

R.1 The differential temperature instruments proposed to be relocated are not assumed to function to mitigate any accident described in Chapters 6 or 15 of the USAR. These differential temperature instruments are provided only to detect and initiate isolation of a 25-gpm-equivalent steam leak. However, these instruments constitute only one method of determining steam leakage in their respective areas. In addition to the temperature monitoring, excess reactor coolant leakage can be detected by low reactor water level, high process line flow, high differential flow, and various other plant specific methods. Several of the BWR-6s have performed studies and analyses which support the relocation of the differential temperature instruments from the Technical Specifications.

The differential temperature instruments are neither used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

The differential temperature isolation instruments are neither used for, nor capable of, monitoring a process variable that is an initial condition of a DBA or transient analyses.

The differential temperature isolation instruments are not used as part of the primary success path in the mitigation of a DBA or transient. No pressure-temperature analyses, radiation dose calculations, or equipment qualification parameters take credit for operation of these differential temperature instruments. In addition, adequate redundancy is available to perform their functions by other methods.

Although the overall isolation instrumentation Function satisfies Criterion 3 of the NRC's Final Policy Statement on Technical Specification Improvement, these differential temperature instruments are not assumed to function to mitigate any DBA or transient.

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Attachment 2
Page 10

Table 3.3.7.1-1 (page 1 of 1)
Control Room Emergency Recirculation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low Low Low, Level 1	1,2,3, (a)	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4 SR 3.3.7.1.5	≥ 14.3 inches
2. Drywell Pressure - High	1,2,3	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4 SR 3.3.7.1.5	≤ 1.88 psig
3. Control Room Ventilation Radiation Monitor	1,2,3, (b)	1	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.4 SR 3.3.7.1.5	≤ 800 cpm

(a) During operations with a potential for draining the reactor vessel.

(b) During CORE ALTERATIONS, or operations with a potential for draining the reactor vessel, and or movement of irradiated fuel assemblies in the primary containment or fuel handling building.

3.3.7.1
C.32

#2

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 The reactor coolant system recirculation loop(s) shall be in operation ~~with the total core flow greater than or equal to 45% of rated core flow, or THERMAL POWER less than or equal to the limit specified in Figure 3.4.1.1-1 and either:~~

a. Two recirculation loops operating, ~~or~~ with matched flows

b. A single recirculation loop operating with the ~~following~~ required limits and ~~conditions:~~ Setpoint modifications.

1. ~~a) The MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit adjusted for single recirculation loop operation per Specification 2.1.2, and~~

~~b) The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits adjusted for single recirculation loop operation per the CORE OPERATING LIMITS REPORT in accordance with Specification 3.2.1, and~~

~~c) The Average Power Range Monitor (APRM) Scram and Rod Block Trip Setpoint and Allowable Value equations adjusted to those values applicable for single recirculation loop operation** per Specifications 2.2.1 and 3.3.6, and~~

~~2. A volumetric recirculation loop drive flow less than or equal to 48,500 gpm, and~~ Total core flow and Thermal Power within limits;

~~3. The recirculation flow control system in the Loop Manual (Position Control) mode, and~~

~~4. THERMAL POWER less than or equal to 2500 Megawatts-thermal.~~

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

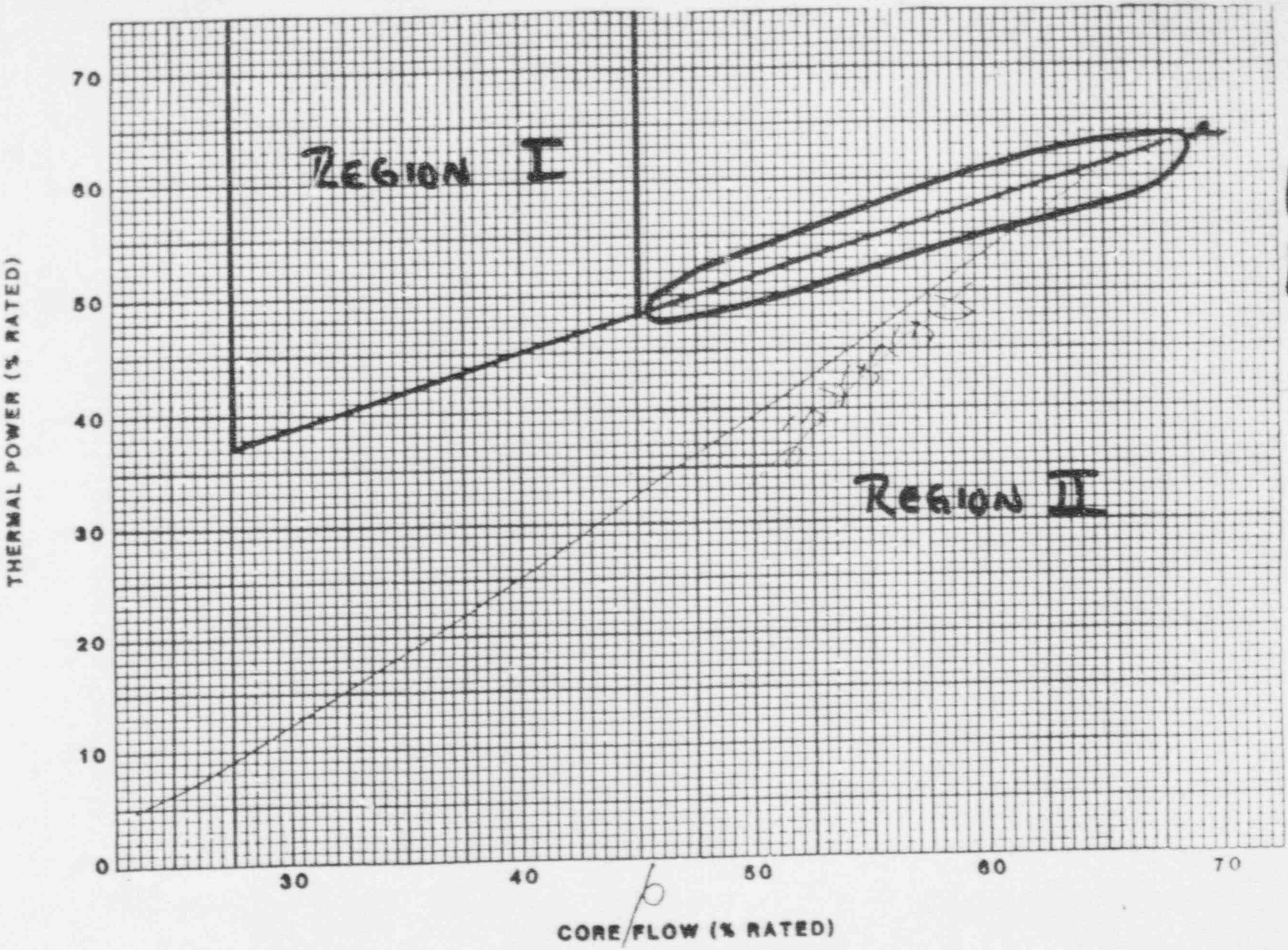
ACTION:

a. Upon initial entry into single loop operation, adjustments to the limits and setpoints of Specifications 2.1.2, 2.2.1**, 3.2.1, and 3.3.6** shall be implemented within 8 hours, or declare the associated equipment inoperable (or declare the associated limits to be "not satisfied"), and take the ACTIONS required by the applicable specifications.

* See Special Test Exception 3.10.4.

** To functionally implement these protective functions during entry into single loop operation, APRM gain adjustments may be made in lieu of adjusting the APRM Scram and Rod Block Flow Biased Setpoints for an interim period of 72 hours.

INSERT



THERMAL POWER VERSUS CORE FLOW

FIGURE 3.4.1.1-1

FIGURE 3.4.1-1

#3

3.4 C.1

LAT

DISCUSSION OF CHANGES
CTS: 3.4.1.1 - RECIRCULATION LOOPS

ADMINISTRATIVE
(continued)

A.6 The specified changes were proposed in a letter to the NRC, PY-CEI/NRR-1353 L, dated June 28, 1991. Due to the extensive changes requested in that letter, Attachment 3A of the submittal letter is inserted into this proposal and then revised accordingly. Footnote ** to that change request is not being carried forward into this change request. The No Significant Hazards Consideration proposed in the letter are still valid. Therefore the change is considered administrative for purposes of this submittal.

A.7 This comment number is not used for this station.

A.8 The existing action to "immediately initiate action to reduce...within 2 hours" is proposed to be revised to "initiate action to restore ...Immediately". The existing requirement would appear to provide a two hours in which the power/flow ratio could exceed the limits, even if capable of being returned to within limits. Also, if the parameters are incapable of being restored to within the limits within 2 hours, the existing action would appear to result in the requirement for an LER. The intent of the action is believed to be more appropriately presented in proposed Required Action D.1. This interpretation of the intent is supported by the BWR Standard Technical Specification, NUREG-1434. As an enhanced presentation of the existing intent, the proposed change is deemed to be administrative.

A.9 The footnote explaining that the numbers given are preliminary and could be changed by a Technical Specification change is proposed to be deleted. This footnote is unnecessary since a Technical specification change can always be submitted if new data determines it is warranted. The footnote does not contain any technical data necessary for the plant operation.

A.10 This comment number is not used for this station.

A.11 A new SR is added to periodically check that the LCO requirements are being met. Since the LCO must always be met, and the operator maintains awareness of the reactor power and recirculation flow parameters, this change is considered administrative in nature.

#3 Insert 2c

RELOCATED SPECIFICATIONS

None in this section.

d3

DISCUSSION OF CHANGES
CTS: 3.4.1.1 - RECIRCULATION LOOPS

(Insert 2a)

A.12 The current LCO description of the allowed region of operation, i.e., "... with the total core flow greater than or equal to 45% of rated core flow, or THERMAL POWER less than or equal to the limit specified in Figure 3.4.1.1-1 ..." is now graphically represented on ITS Figure 3.4.1-1 as Region II. Since these regions are equivalent, the proposed change is deemed to be administrative.

REACTOR COOLANT SYSTEM

LCO 3.4.5

OPERATIONAL LEAKAGE

LCO 3.4.6

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

LCO 3.4.5

a. No PRESSURE BOUNDARY LEAKAGE.

INSERT LCO 3.4.5.d (M1)

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 25 gpm IDENTIFIED LEAKAGE averaged over any 24-hour period. (LA1)

SR 3.4.6.1 d. 0.5 gpm leakage per nominal inch of valve size up to a maximum of 5 gpm from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1, ~~at rated pressure.~~ (L2)

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3. ← INSERT LCO 3.4.6 MODE 3 PIV EXCEPTION FOR RHR SDC Flowpath

ACTION:

LCO 3.4.5 COND C a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

LCO 3.4.5 COND A COND C b. With any reactor coolant system leakage greater than the limits in b and/or c, above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

LCO 3.4.6 COND A COND B c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one other closed manual or deactivated automatic or check* valve, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. (LA5) R.A. A Note — 3.4 C.7c (#4)

INSERT LCO 3.4.6 ACTIONS NOTE 1 (A1)

INSERT LCO 3.4.6 ACTIONS NOTE 2 (A2)

INSERT LCO 3.4.5 COND B (M1)

(LA1)

*Which have been verified not to exceed the allowable leakage limit at the last refueling outage or after the last time the valve was disturbed, whichever is more recent.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Pressure Isolation Valve (PIV) Leakage

LCO 3.4.6 The leakage from each RCS PIV shall be within limit.

APPLICABILITY: MODES 1 and 2,
MODE 3, except valves in the residual heat removal (RHR) shutdown cooling flowpath ~~are not required to meet the requirements of this LCO~~ when in the shutdown cooling mode of operation.

C1

, or during the transition to or from,

ACTIONS

---NOTES---

1. Separate Condition entry is allowed for each flow path.
2. Enter applicable Conditions and Required Actions for systems made inoperable by PIVs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
C2 A. Leakage from one or more RCS PIVs not within limit. One or more flow paths with	---NOTE--- Each valve used to satisfy Required Action A.1 and Required Action A.2 shall have been verified to meet SR 3.4.6.1 and be in the reactor coolant pressure boundary [or the high pressure portion of the system].	a ~~C1~~
		(continued)

--- check NOTE ---
Each valve used to satisfy Required Action A.1 shall have been verified to meet SR 3.4.6.1 and be in the reactor coolant pressure boundary or in the high pressure portion of the system.

#4 3.4 C:70

BASES (continued)

APPLICABILITY

3.4 C.80

In MODES 1, 2, and 3, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 3, valves in the RHR flowpath are not required to meet the requirements of this LCO when in, or during transition to or from, the RHR Shutdown Cooling mode of operation.

C8

In MODES 4 and 5, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment. Accordingly, the potential for the consequences of reactor coolant leakage is far lower during these MODES.

ACTIONS

The ACTIONS are modified by two Notes. Note 1 has been provided to modify the ACTIONS related to RCS PIV flow paths. Section 1.3, Completion Times, specifies once a Condition has been entered, subsequent ~~trains~~ divisions, subsystems, components or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for the Condition of RCS PIV leakage limits exceeded provide appropriate compensatory measures for separate affected RCS PIV flow paths. As such, a Note has been provided that allows separate Condition entry for each affected RCS PIV flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system OPERABILITY, or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function. As a result, the applicable Conditions and Required Actions for systems made inoperable by PIVs must be entered. This ensures appropriate remedial actions are taken, if necessary, for the affected systems.

C5

A.1 ~~and A.2~~ P5

#4

If leakage from one or more RCS PIVs is not within limit, the flow path must be isolated by at least one closed manual, deactivated ~~automatic~~, or check valve within 4 hours. ~~Required Action A.1 and Required Action A.2 are~~

C4

~~INSERT B28A~~

3.4 C.70

(continued)

BASES

ACTIONS

A.1 ~~and A.2~~ (continued)

Required Action A.1 is ~~modified by a Note stating that the valves used for isolation must meet the same leakage requirements as the PIVs and must be on the RCPB for the high pressure portion of the system.~~ INSERT B28A

Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the flow path if leakage cannot be reduced while corrective actions to reseat the leaking PIVs are taken. The 4 hours allows time for these actions and restricts the time of operation with leaking valves.

~~Required Action A.2 specifies that the double isolation barrier of two valves be restored by closing another valve qualified for isolation or restoring one leaking PIV. The 72 hour Completion Time after exceeding the limit considers the time required to complete the Required Action, the low probability of a second valve failing during this time period, and the low probability of a pressure boundary rupture of the low pressure ECCS piping when overpressurized to reactor pressure (Ref. 8).~~

B.1 and B.2

If leakage cannot be reduced or the system isolated, 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 and to MODE 4 within 36 hours. This action may reduce the leakage and also reduces the potential for a LOCA outside the containment. The Completion Times are reasonable, based on operating experience, to achieve the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

Performance of leakage testing on each RCS PIV is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable

(continued)

INSERT B28A

A check valve may be used for this purpose if leakage past the check valve did not exceed the allowable leakage limit at the last refueling outage, or after the last time the valve was known to have opened, whichever is more recent.

#4 - unchanged from earlier submittal

DISCUSSION OF CHANGES TO NUREG-1434
TS 3.4.6 - RCS PRESSURE ISOLATION VALVE LEAKAGE

CHANGE/IMPROVEMENT TO NUREG STS
(continued)

C.5 These changes are proposed to revise specific terminology to that which is generically preferred for application to the BWR/6 plants. The BWR LCOs do not use the term "train", however, "division" is used in several places.

C.6 This change is to provide clarity and prevent including a list containing excessive procedural type details in the Required Action of the Technical Specifications.

C.7 Note deleted from Required Action and appropriate stipulations from NOTE added to Bases. This is consistent with this kind of statement made other places in Technical Specifications.

C.8 Correct mode of RHR system operations inserted into Bases Applicability statement.

C.9 The reference is deleted from the text since the limit being discussed could not be identified in the reference.

Comment not used at this station.

3.4
C.70

#4

<No Separate LCO> -- A1

LCO 3.6.1.1 / LCO 3.6.1.3 #5 3.6.1.1 C:6

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

A1 ~~3.6.1.2~~ Primary containment leakage rates shall be limited to: A1

SR 3.6.1.1.1 and a. An overall integrated leakage rate of less than or equal to 0.75 L_a, 0.20 percent by weight of the primary containment air per 24 hours at P_a, 11.31 psig. A4

SR 3.6.1.2.1 b. A combined leakage rate of less than or equal to 0.60 L_a for all penetrations and all valves, except for main steam line isolation valves and valves which are hydrostatically leak tested, subject to Type B and C tests when pressurized to P_a, 11.31 psig. A4 A1 S:3.6 3.6.1.3 C:39

SR 3.6.1.3.9 c. 10 Less than or equal to 25 scf per hour for any one main steam line through the isolation valves when tested at P_a, 11.31 psig. A4 STET L9

d. A combined leakage rate of less than or equal to 0.0504 L_a for all

SR 3.6.1.3.8 9 penetrations that are secondary containment bypass leakage paths when pressurized to the required test pressure. S:3.6 3.6.1.3 C:29

SR 3.6.1.3.10 e. 11 A combined leakage rate of less than or equal to 1 gpm times the total number of containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at 1.10 P_a, 12.44 psig. A4

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3. A5

ACTION: A1 { LCO 3.6.1.1
LCO 3.6.1.3 ACTION NOTE 4

With:

a. The measured overall integrated primary containment leakage rate exceeding 0.75 L_a, or A1

b. The measured combined leakage rate for all penetrations and all valves except for main steam line isolation valves and valves which are hydrostatically leak tested, subject to Type B and C tests exceeding 0.60 L_a, or A4 STET L9

c. The measured leakage rate exceeding 25 scf per hour for any one main steam line through the isolation valves, or S:3.6 3.6.1.3 C:39

d. The combined leakage rate for all penetrations that are secondary containment bypass leakage paths exceeding 0.0504 L_a, or

e. The measured combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment exceeding 1 gpm times the total number of such valves:

A5

*See Special Test Exception 3.10.1.

DISCUSSION OF CHANGES
CTS: 3.6.1.2 - PRIMARY CONTAINMENT LEAKAGE #5 3.6.1.1 C.6

ADMINISTRATIVE
(continued)

This comment number is not used for this station.

A.3 ~~These changes have already been proposed to the NRC in a letter, PY-CEI/NRR-1576 L dated March 1, 1993. The .75 proposed to be deleted from LCO 3.6.1.2 a. is merely a different presentation of what has been proposed in the letter to the NRC, In that request it was proposed, among other changes, to insert wording into this requirement to clarify that the requirement was only for as-left leakage rates. Therefore as found leakage rates up to La are acceptable. The proposed change made in this present submittal, presents a clearer approach, but still contains the exact same requirements, that a leak rate of La is acceptable as found, but has to be made less than or equal to .75 La as-left. The No Significant Hazards Consideration written as part of that submittal is still valid. Therefore this change is considered administrative.~~

A.4 The deletion of these numerical values were already proposed in a letter to the NRC, PY-CEI/NRR-1510 L dated June 24, 1992. The No Significant Hazards Consideration written as part of that submittal is still valid. Therefore this change is considered administrative.

A.5 The format of the proposed Technical Specifications does not include providing "cross references." The existing reference to "See Special Test Exception 3.10.1" serves no functional purpose, and therefore its removal is purely an administrative difference in presentation.

A.6 This comment number is not used for this station.

A.7 This comment number is not used for this station.

A.8 This comment number is not used for this station.

RELOCATED SPECIFICATIONS

None in this section.

REACTOR COOLANT SYSTEM

SQ 3.4.11.3
& SQ 3.4.11.4

IDLE RECIRCULATION LOOP STARTUP

LIMITING CONDITION FOR OPERATION

SQ 3.4.11.3

~~3.4.1.4~~ An idle recirculation loop shall not be started unless the temperature differential between the reactor pressure vessel steam space coolant and the bottom head drain line coolant is less than or equal to 100°F*, and: Select #6 ~~LA1~~

SQ 3.4.11.4 A1

a. When both loops have been idle, unless the temperature differential between the reactor coolant within the idle loop to be started up and the coolant in the reactor pressure vessel is less than or equal to 50°F, or Select

b. When only one loop has been idle, unless the temperature differential between the reactor coolant within the idle and operating recirculation loops is less than or equal to 50°F, and the operating loop flow rate is less than or equal to 50% of rated loop flow.

SQ 3.4.11.3 Note
SQ 3.4.11.4 Note
LA2

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

ACTION:

With temperature differences and/or flow rates exceeding the above limits, suspend startup of any idle recirculation loop.

SURVEILLANCE REQUIREMENTS

SQ 3.4.11.3
SQ 3.4.11.4

LA2

~~4.4.1.4~~ The temperature differentials and flow rate shall be determined to be within the limits within 15 minutes prior to startup of an idle recirculation loop.

SQ 3.4.11.3 Note

*Below 25 psig, this temperature differential is not applicable.

DISCUSSION OF CHANGES
CTS: 3.4.1.4 - IDLE RECIRCULATION LOOP STARTUP

ADMINISTRATIVE

A.1 Thermal stresses on vessel components during recirculation loop startups are dependent on the temperature difference between the idle loop coolant and the RPV coolant. Proposed SR 3.4.11.4 ensures the temperature difference between any loop to be started and the RPV coolant is acceptable. A requirement to monitor the temperature difference between an idle loop and an operating loop is unnecessary and can be deleted as it is redundant to the loop-to-coolant requirement of SR 3.4.11.4. However, the loop-to-coolant temperature check may use the operating loop temperature as representative of "coolant temperature."

RELOCATED SPECIFICATIONS

None in this section.

TECHNICAL CHANGE - MORE RESTRICTIVE

None in this section.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic" Not used at this station. #6

LA.1 ~~The details relating to RPV pressure and temperature limits have been relocated to the Pressure and Temperature Limits Report (PTLR). These limits for system operation are also described in the USAR. Changes to the PTLR will be controlled by the provisions of the proposed Reporting Requirements section in Chapter 5 of the Technical Specifications.~~

LA.2 The flow rate of an operating loop during idle loop startup is not a required initial condition of any transient analysis. The speed requirement is an operational limit to reduce the probability of scram during idle loop startup and is therefore covered by plant specific procedures.

"Specific"

None in this section.

DISCUSSION OF CHANGES
CTS: 3.8.2.1 - D.C. SOURCES - OPERATING

ADMINISTRATIVE
(continued)

A.8 The present wording changed to clarify that inter-cell connections also include inter-rack and inter-tier connections where required. Since this does not represent any change to present requirements, but is merely a clarification, the change is considered administrative.

RELOCATED SPECIFICATIONS

None in this section.

TECHNICAL CHANGE - MORE RESTRICTIVE

M.1 #7 72 3.8 C:93 In the event a battery experiences a discharge (not a momentary transient as in the starting of a large load) which brings the battery terminal voltage below 110 volts, confirmation of continued battery OPERABILITY should be made sooner than 7 days. The proposed change to 24 hours allows sufficient time to plan for an unscheduled surveillance and complete the performance of the surveillance of cell parameters without undo haste. Since this confirmation of OPERABILITY is revised to make that determination sooner, the change is conservative.

M.2 The allowance to correct the Category B limit for temperature is being proposed for deletion based on the BWR Standard Technical Specification, NUREG-1434, presentation and IEEE-450 recommendations.

M.3 Limitations are proposed to be imposed on this allowance: the utilization of charging current limited to 7 days, and a requirement to measure actual specific gravities at the end of this period. These restrictions will assure excessive reliance on charging current is not made.

M.4 Proposed Required Action A.1 requires a more immediate check that pilot cell electrolyte level and float voltage are within limits. Required Action A.2 proposes a periodic re-verification that all cell parameters are within limits. These restrictions provide added assurance of adequate battery capabilities for the period allowed to completely restore the cell parameters.

M.5 The frequency for Surveillance 4.8.2.1.f has been changed from 18 months to 12 months. Since the change will require the test on a more frequent bases, the change is considered more restrictive.

Summary of Perry Nuclear Power Plant
Totally and Partially Relocated Technical Specifications

Existing TS	Title	Relocation Document	Relocation Control
1.2	Operational Conditions	ORM/Bases	§50.59
4.0.5	Inservice Inspection and Testing Programs	ORM	§50.55a §50.59
3/4.1.2	Reactivity Anomalies	ORM/Bases	§50.59
3/4.1.3.1	Control Rod Operability	ORM/Bases	§50.59
3/4.1.3.2	Control Rod Maximum Scram Insertion Times	ORM/Bases	§50.59
3/4.1.3.3	Control Rod Scram Accumulators	ORM/Bases	§50.59
3/4.1.3.4	Control Rod Drive Coupling	ORM/Bases	§50.59
3/4.1.3.5	Control Rod Position Indication	ORM/Bases	§50.59
3/4.1.3.6	Control Rod Drive Housing Support	ORM/Bases	§50.59
3/4.1.4.1	Control Rod Withdrawal	ORM/Bases	§50.59
3/4.1.4.2	Rod Pattern Control System	ORM/Bases	§50.59
3/4.1.5	Standby Liquid Control System	ORM/Bases	§50.59
3/4.2.1	Average Planar Linear Heat Generation Rate	USAR/Bases	§50.59
3/4.2.2	Mininum Critical Power Ratio	USAR/Bases	§50.59
3/4.2.3	Linear Heat Generation Rate	USAR/Bases	§50.59
3/4.3.1	Reactor Protection System Instrumentation	ORM/Bases	§50.59
3/4.3.2	Isolation Actuation Instrumentation	ORM/Bases	§50.59
3/4.3.3	Emergency Core Cooling System Actuation Instrumentation	ORM/Bases	§50.59
3/4.3.4.1	ATWS-RPT System Instrumentation	ORM/Bases	§50.59
3/4.3.4.2	EOC-RPT System Instrumentation	ORM/Bases	§50.59
3/4.3.5	RCIC System Instrumentation	ORM/Bases	§50.59
3/4.3.6	Control Rod Block Instrumentation	ORM/Bases	§50.59
3/4.3.7.1	Radiation Monitoring Instrumentation	ORM/Bases	§50.59
3/4.3.7.2	Seismic Monitoring Instrumentation	ORM/USAR	§50.59
3/4.3.7.3	Meteorological Monitoring Instrumentation	ORM/USAR	§50.59
3/4.3.7.4	Remote Shutdown Instrumentation	ORM/Bases	§50.59
3/4.3.7.5	Accident Monitoring Instrumentation	ORM/Bases	§50.59
3/4.3.7.6	Source Range Monitors	ORM/Bases	§50.59
3/4.3.7.7	Traversing In-Core Probe System	ORM/USAR	§50.59

Existing TS	Title	Relocation Document	Relocation Control
3/4.3.7.8	Loose-Part Detection System	ORM/USAR	§50.59
3/4.3.7.9	Radioactive Liquid Effluent Monitoring Instrumentation	ORM/USAR	§50.59
3/4.3.7.10	Radioactive Gaseous Effluent Monitoring Instrumentaion	ORM/USAR	§50.59
3/4.3.8	Turbine Overspeed Protection System	ORM/USAR	§50.59
3/4.3.9	Plant Systems Actuation Instrumentation	ORM/USAR/Bases	§50.59
3/4.4.1.1	Recirculation Loops	ORM/Bases	§50.59
3/4.4.1.4	Idle Recirculation Loop Startup	ORM/Bases	§50.59
3/4.4.2.1	Safety/Relief Valves	ORM/Bases	§50.59
3/4.4.2.2	S/RVs Low-Low Set Function	ORM/Bases	§50.59
3/4.4.3.2	Operational Leakage	ORM/Bases	§50.59
3/4.4.4	Chemistry	ORM/Bases	§50.59
3/4.4.7	Main Steam Line Isolation Valves	ORM/Bases	§50.59
3/4.4.8	Structural Integrity	USAR	§50.59
3/4.4.9.1	RHR Hot Shutdown	ORM/Bases	§50.59
3/4.4.9.2	RHR Cold Shutdown	ORM/Bases	§50.59
3/4.5.1	ECCS-Operating	ORM/Bases	§50.59
3/4.5.2	ECCS - Shutdown	ORM/Bases	§50.59
3/4.5.3	Suppression Pool	ORM/Bases	§50.59
3/4.6.1.1.1	Primary Containment Integrity - Operating	ORM/Bases	§50.59
3/4.6.1.1.2	Primary Containment Integrity - Shutdown	USAR/Bases	§50.59
3/4.6.1.3	Containment Air Locks	ORM/Bases	§50.59
3/4.6.1.4	Main Steam Isolation Valve Leakage Control System	ORM/Bases	§50.59
3/4.6.1.7	Primary Containment Average Air Temperature	USAR/Bases	§50.59
3/4.6.1.9	Feedwater Leakage Control System	USAR/Bases	§50.59
3/4.6.2.2	Drywell Bypass Leakage	ORM/Bases	§50.59
3/4.6.2.3	Drywell Air Lock	ORM/Bases	§50.59
3/4.6.2.4	Drywell Structural Integrity	ORM/Bases	§50.59
3/4.6.2.6	Drywell Average Air Temperature	ORM/Bases	§50.59
3/4.6.3.1	Suppression Pool	ORM/Bases	§50.59
3/4.6.3.2	Containment Spray	ORM/Bases	§50.59

Existing TS	Title	Relocation Document	Relocation Control
3/4.6.3.3	Suppression Pool Cooling	ORM/Bases	§50.59
3/4.6.3.4	Suppression Pool Makeup	ORM/Bases	§50.59
3/4.6.4	Containment Isolation Valves	ORM/Bases	§50.59
3/4.6.5.1	Containment Vacuum Breakers	USAR/Bases	§50.59
3/4.6.5.2	Containment Humidity Control	USAR/Bases	§50.59
3/4.6.5.3	Drywell Vacuum Breakers	USAR/Bases	§50.59
3/4.6.6.1	Secondary Containment Integrity	ORM/Bases	§50.59
3/4.6.6.2	Annulus Exhaust Gas Treatment System	USAR/Bases	§50.59
3/4.6.7.1	Containment Hydrogen Recombiner Systems	ORM/Bases	§50.59
3/4.6.7.2	Combustible Gas Mixing System	ORM/Bases	§50.59
3/4.6.7.3	Containment and Drywell Hydrogen Ignition System	ORM/Bases	§50.59
3/4.7.1.1	Emergency Service Water System (Loops A, B, C)	ORM/Bases	§50.59
3/4.7.1.2	Emergency Closed Cooling Water System	ORM/Bases	§50.59
3/4.7.2	Control Room Emergency Recirculation System	ORM/Bases	§50.59
3/4.7.3	Reactor Core Isolation Cooling System	USAR/Bases	§50.59
3/4.7.4	Snubbers	ORM	§50.59
3/4.7.5	Sealed Source Contamination	ORM	§50.59
3/4.7.6	Main Turbine Bypass System	ORM/Bases	§50.59
3/4.7.7.1	Fuel Handling Building Ventilation System	ORM/Bases	§50.59
3/4.7.7.2	Fuel Handling Building (FHB) Integrity	ORM/Bases	§50.59
3/4.8.1.1	AC Sources - Operating	ORM/Bases	§50.59
3/4.8.1.2	AC Sources - Shutdown	ORM/Bases	§50.59
3/4.8.2.1	DC Sources - Operating	ORM/Bases	§50.59
3/4.8.2.2	DC Sources - Shutdown	ORM/Bases	§50.59
3/4.8.3.1	Onsite Power Distribution Systems - Operating	ORM/Bases	§50.59
3/4.8.3.2	Onsite Power Distribution Systems - Shutdown	ORM/Bases	§50.59
3/4.8.4.1	Containment Penetration Conductor Overcurrent Protective Devices	ORM/Bases	§50.59
3/4.9.1	Reactor Mode Switch	ORM/Bases	§50.59
3/4.9.2	Instrumentation	ORM/Bases	§50.59
3/4.9.4	Decay Time	ORM	§50.59

Existing TS	Title	Relocation Document	Relocation Control
3/4.9.5	Communications	ORM	§50.59
3/4.9.6	Refueling Platform	ORM	§50.59
3/4.9.7	Crane Travel	ORM	§50.59
3/4.9.8	Water Level - Reactor Vessel	ORM/Bases	§50.59
3/4.9.9	Water Level-Spent Fuel Storage and Upper Containment Fuel Pools	USAR/Bases	§50.59
3/4.9.10.1	Single Control Rod Removal	ORM/Bases	§50.59
3/4.9.10.2	Multiple Control Rod Removal	ORM/Bases	§50.59
3/4.9.11.1	Residual Heat Removal and Coolant Circulation - High Water Level	ORM/Bases	§50.59
3/4.9.11.2	Residual Heat Removal and Coolant Circulation - Low Water Level	ORM/Bases	§50.59
3/4.9.12	Inclined Fuel Transfer System	ORM	§50.59
3/4.11.1.1	Liquid Effluents	USAR	§50.59
3/4.11.1.2	Dose	USAR	§50.59
3/4.11.1.3	Liquid Radwaste Treatment System	USAR	§50.59
3/4.11.1.4	Liquid Holdup Tanks	USAR	§50.59
3/4.11.2.1	Dose Rate	USAR	§50.59
3/4.11.2.2	Dose - Noble Gases	USAR	§50.59
3/4.11.2.3	Dose - I-131, I-133, Tritium and Radionuclides in Particulate Form	USAR	§50.59
3/4.11.2.4	Gaseous Radwaste (Offgas) Treatment	USAR	§50.59
3/4.11.2.5	Ventilation Exhaust Treatment Systems	ORM/Bases	§50.59
3/4.11.2.6	Explosive Gas Mixture	ORM/Bases	§50.59
3/4.11.2.7	Main Condenser	ORM/Bases	§50.59
3/4.11.3	Solid Radwaste Treatment	ORM/Bases	§50.59
3/4.11.4	Total Dose	ORM/Bases	§50.59
3/4.12.1	Monitoring Program	ORM/Bases	§50.59
3/4.12.2	Land Use Census	ORM/Bases	§50.59
3/4.12.3	Interlaboratory Comparison Program	ORM/Bases	§50.59
5.1.1	Exclusion Area, Unrestricted Area for Liquid Effluents, and Site Boundary for Gaseous Effluents	ORM/USAR	§50.59
5.2.1	Containment Configuration	ORM/USAR	§50.59
5.2.2	Design Temperature and Pressure	ORM/USAR	§50.59

Existing TS	Title	Relocation Document	Relocation Control
5.2.3	Secondary Containment	ORM/USAR	§50.59
5.3.2	Control Rod Assemblies	ORM/USAR	§50.59
5.4.1	Reactor Coolant System Design Pressure and Temperature	ORM/USAR	§50.59
5.4.2	Reactor Coolant System Volume	ORM/USAR	§50.59
5.5.1	Meteorological Tower Location	ORM/USAR	§50.59
5.6.1	Fuel Storage Criticality	ORM/USAR	§50.59
5.7.1	Component Cyclic or Transient Limit	ORM/USAR	§50.59
6.1.2	Line of Authority Directive	ORM	§50.59
6.2.2.a, Table 6.2.2-1, 6.2.2.d	Minimum Shift Crew Composition and SRO Requirements	ORM	§50.54(m) §50.59
6.2.3	Independent Safety Engineering Group	ORM	§50.59
6.2.4.1	Shift Technical Advisor	ORM	§50.59
6.4	Training	ORM	§50.59 §50.55
6.5	Review and Audit	ORM	§50.54(a)
6.6	Reportable Event Action	ORM	§50.59 §50.73
6.7	Safety Limit Violation	ORM	§50.72 §50.73 §50.59
6.8.1.c	Security Plan Implementation	ORM	§50.54(p)
6.8.1.d	Emergency Plan Implementation	ORM	§50.54(q)
6.8.1.e	Process Control Program Implementation	ORM/ODCM	§50.59
6.8.1.g	Radiological Environmental Monitoring Program Implementation	ORM/ODCM	§50.59
6.8.1.h	Fire Protection Program Implementation	ORM	§50.48 §50.59 LC 2.C(6)
6.8.1.i	Regulatory Guide 4.15	ORM	§50.59
6.8.2	Review and Approval Process	ORM	§50.54(a)
6.8.3	Temporary Changes	ORM	§50.59
6.8.4.b	In-Plant Radiation Monitoring	ORM	§50.59 §20
6.9.1.1,2,3	Startup Reports	ORM	§50.59

Existing TS	Title	Relocation Document	Relocation Control
6.9.2	Special Reports	ORM/Bases	§50.4 §50.59 §50.73
6.10	Record Retention	ORM	§50.54(a) §20 §71
6.11	Radiation Protection Program	ORM	§50.59 §20
6.12	High Radiation Area	ORM	§50.59 §20
6.13	Process Control Program	PCP	§50.59 §20 §61 §71
6.14	Offsite Dose Calculation Manual	ODCM	§50.59

ORM - Operational Requirements Manual
USAR - Updated Safety Analysis Report
ODCM - Off-site Dose Calculation Manual
LC 2.C(6) - License Condition 2.C(6)
PCP - Process Control Program