Final Submittal
(Blue Paper)

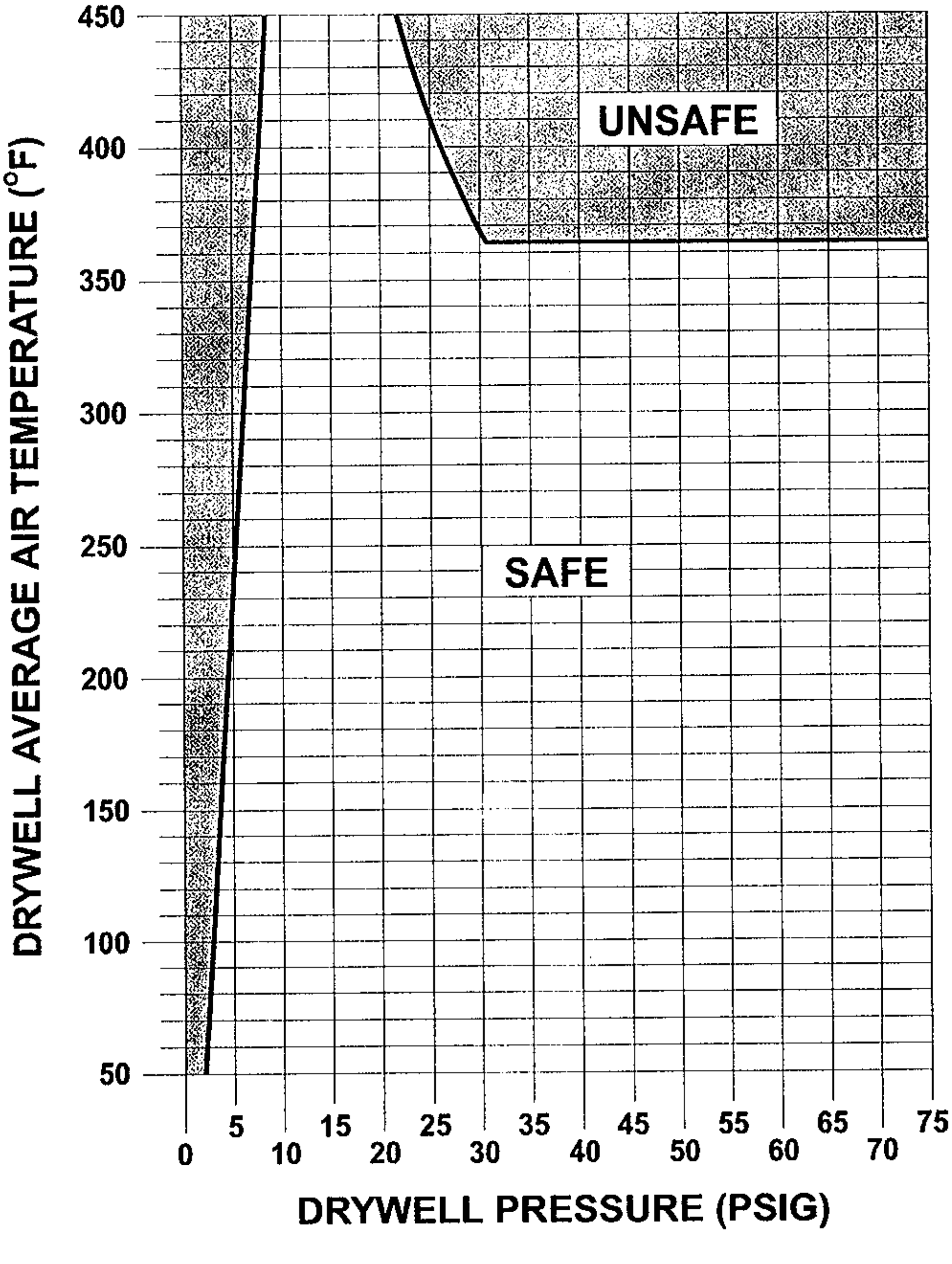
1. Final RO/SRO Written Examination References

BRUNSWICK EXAM
50-2003-301
50-325 & 50-324

FEBRUARY 10 - 13 & 19, 2003

ATTACHMENT 5 (Cont'd)

FIGURE 1
DRYWELL SPRAY INITIATION LIMIT

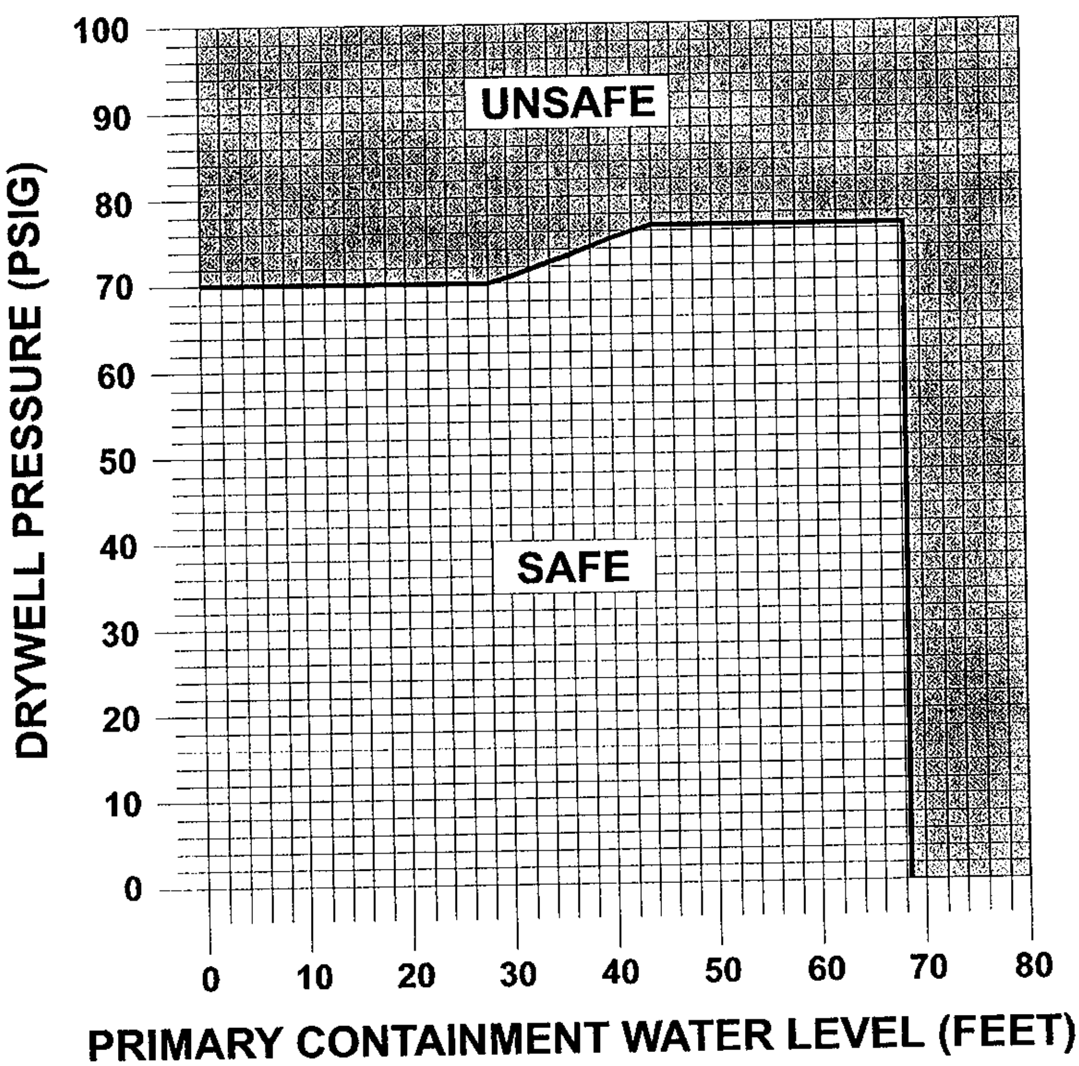


NOTE

DRYWELL AVERAGE AIR TEMPERATURE MAY BE DETERMINED USING ATTACHMENT 4 OF THE "USER'S GUIDE"

ATTACHMENT 5 (Cont'd)

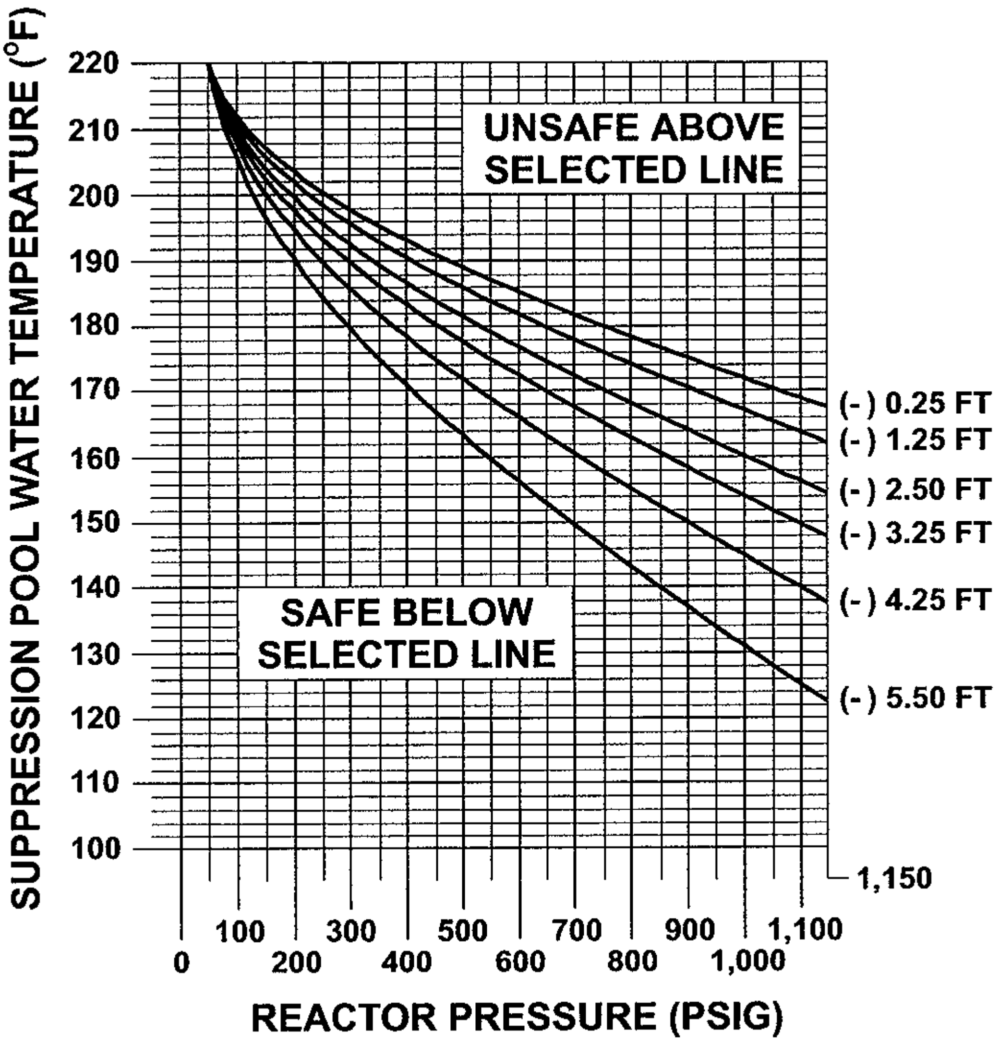
FIGURE 2
PRIMARY CONTAINMENT PRESSURE LIMIT-A



If using the following instrument:	PCPL-A is:
CAC-PI-1230	70 psig
CAC-PI-4176	Use the graph
CAC-PR-1257-1	Use the graph

ATTACHMENT 5 (Cont'd)

FIGURE 3
UNIT 1 HEAT CAPACITY TEMPERATURE LIMIT



NOTE

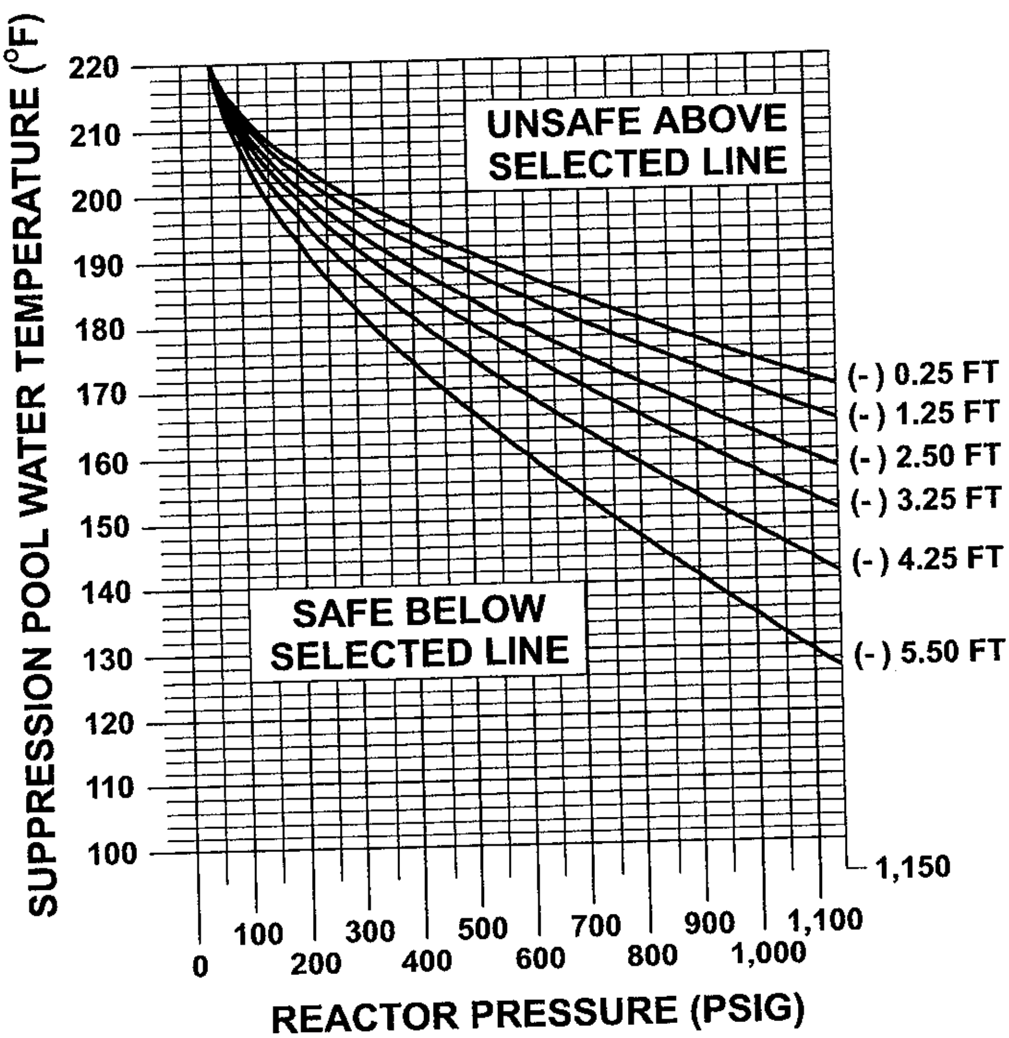
SUPPRESSION POOL WATER TEMPERATURE IS DETERMINED BY:

CAC-TR-4426-1A POINT WTR AVG
OR CAC-TR-4426-2A POINT WTR AVG
OR COMPUTER POINT G050
OR COMPUTER POINT G051
OR CAC-TY-4426-1
OR CAC-TY-4426-2

SELECT GRAPH LINE IMMEDIATELY BELOW SUPPRESSION POOL WATER LEVEL AS THE LIMIT.

ATTACHMENT 5 (Cont'd)

FIGURE 3A
UNIT 2 HEAT CAPACITY TEMPERATURE LIMIT



NOTE

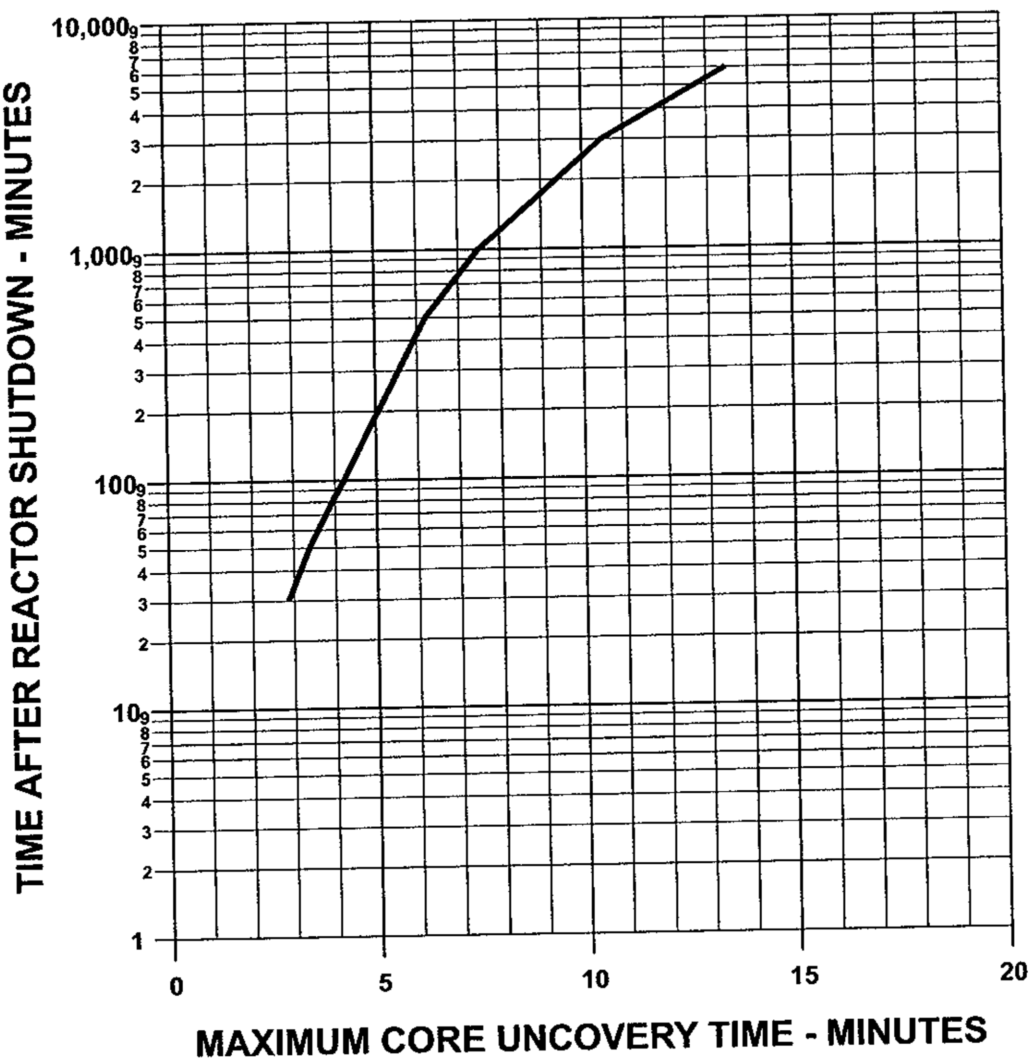
SUPPRESSION POOL WATER TEMPERATURE IS DETERMINED BY:

CAC-TR-4426-1A POINT WTR AVG
OR CAC-TR-4426-2A POINT WTR AVG
OR COMPUTER POINT G050
OR COMPUTER POINT G051
OR CAC-TY-4426-1
OR CAC-TY-4426-2

SELECT GRAPH LINE IMMEDIATELY BELOW SUPPRESSION POOL WATER LEVEL AS THE LIMIT.

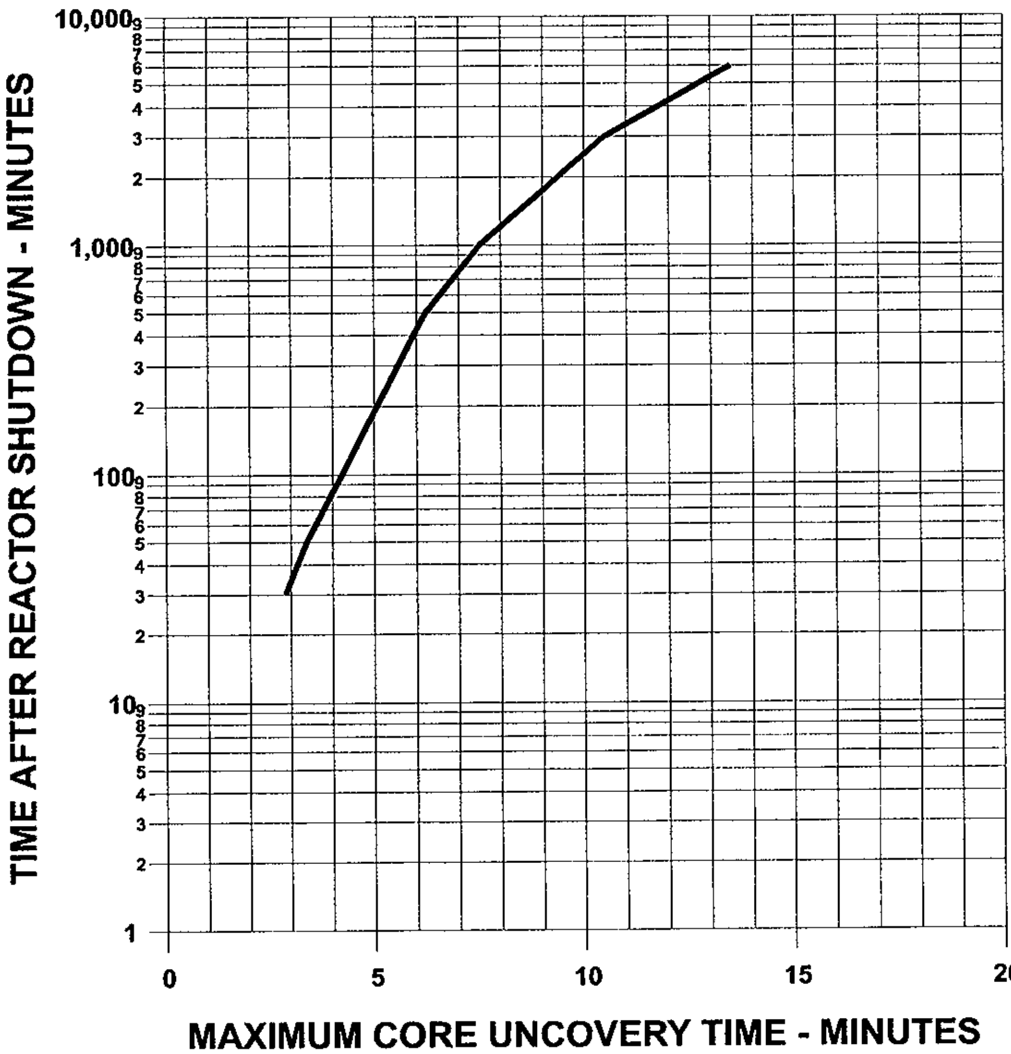
ATTACHMENT 5 (Cont'd)

FIGURE 4
UNIT 1 MAXIMUM CORE UNCOVERY TIME LIMIT



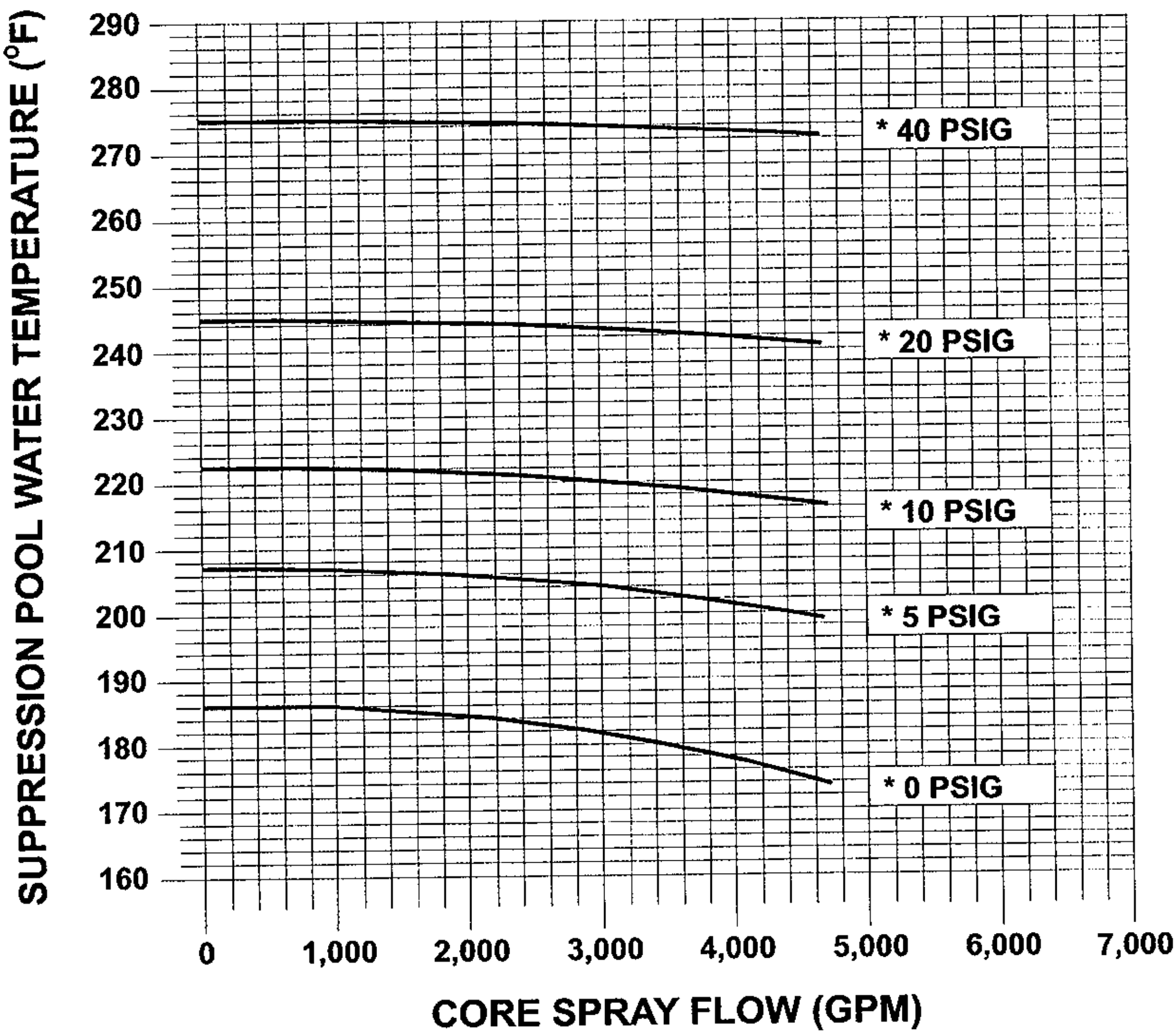
ATTACHMENT 5 (Cont'd)

FIGURE 4A
UNIT 2 MAXIMUM CORE UNCOVERY TIME LIMIT



ATTACHMENT 5 (Cont'd)

FIGURE 5
CORE SPRAY NPSH LIMIT



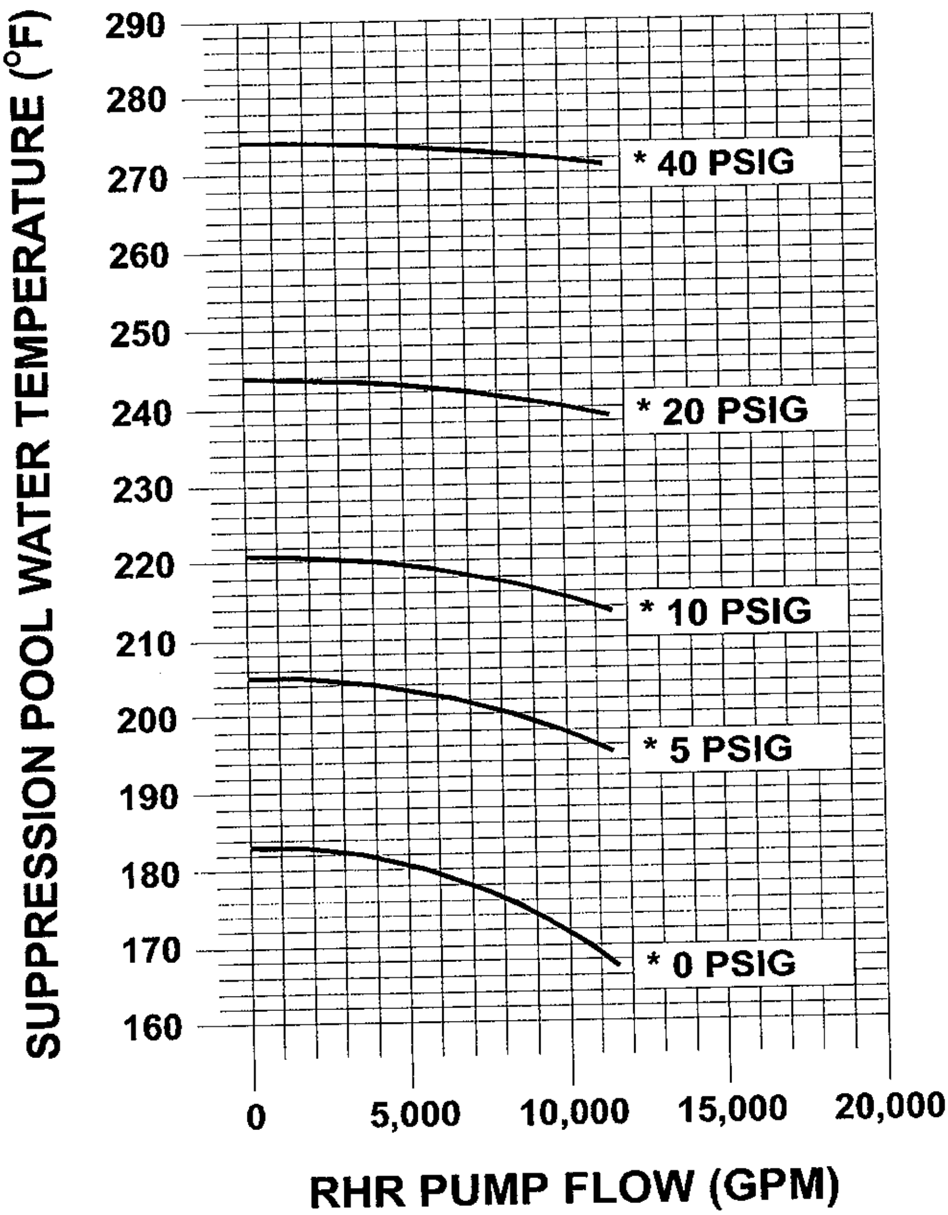
<u>NOTE</u>

SUBTRACT 0.5 PSIG FROM INDICATED SUPPRESSION CHAMBER PRESSURE FOR EACH FOOT OF WATER LEVEL BELOW A SUPPRESSION POOL WATER LEVEL OF -31 INCHES (-2.6 FEET).

*SUPPRESSION CHAMBER PRESSURE (CAC-PI-1257-2A OR CAC-PI-1257-2B)

ATTACHMENT 5 (Cont'd)

FIGURE 6
RHR NPSH LIMIT



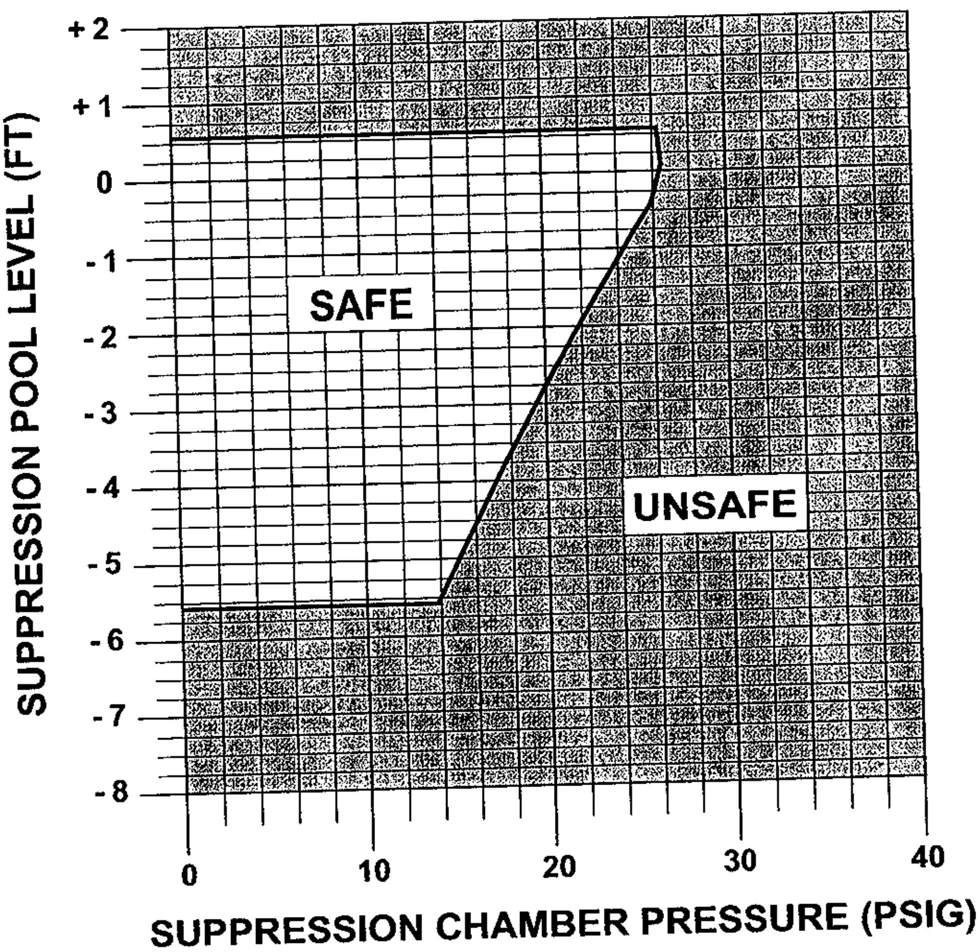
<u>NOTE</u>

SUBTRACT 0.5 PSIG FROM INDICATED SUPPRESSION CHAMBER PRESSURE FOR EACH FOOT OF WATER LEVEL BELOW A SUPPRESSION POOL WATER LEVEL OF -31 INCHES (-2.6 FEET).

*SUPPRESSION CHAMBER PRESSURE (CAC-PI-1257-2A OR CAC-PI-1257-2B)

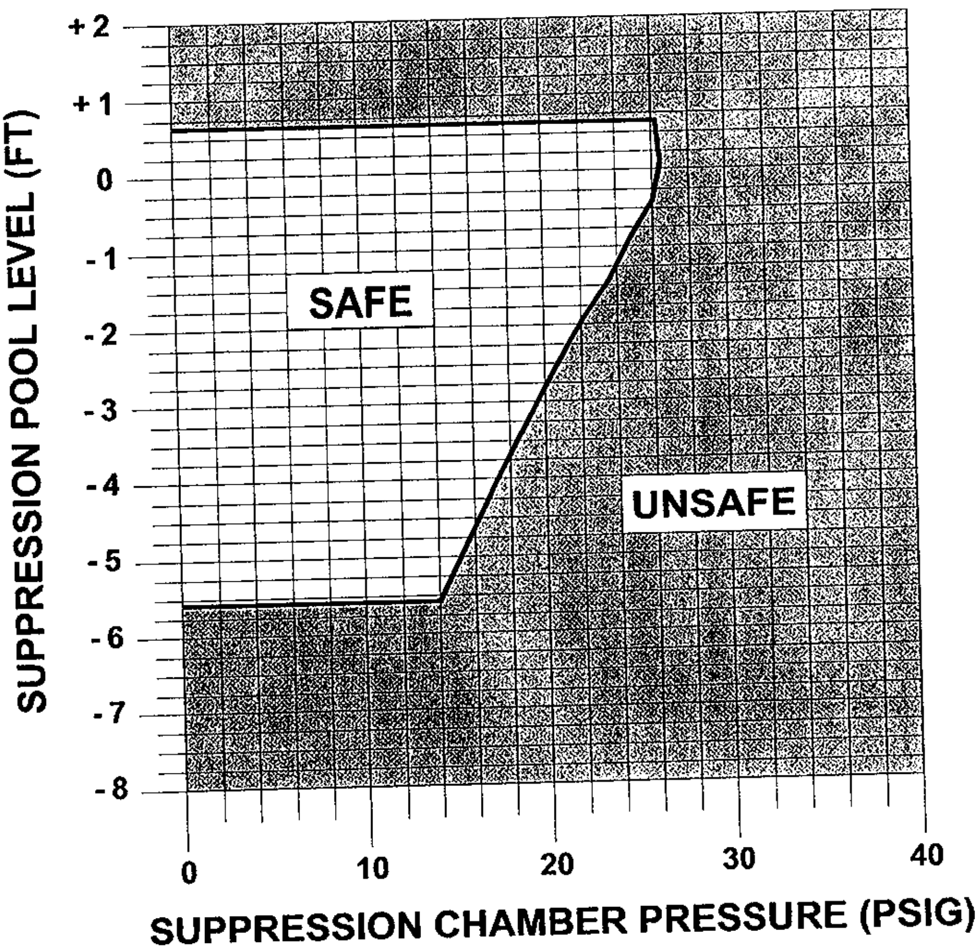
ATTACHMENT 5 (Cont'd)

FIGURE 7
UNIT 1 PRESSURE SUPPRESSION PRESSURE



ATTACHMENT 5 (Cont'd)

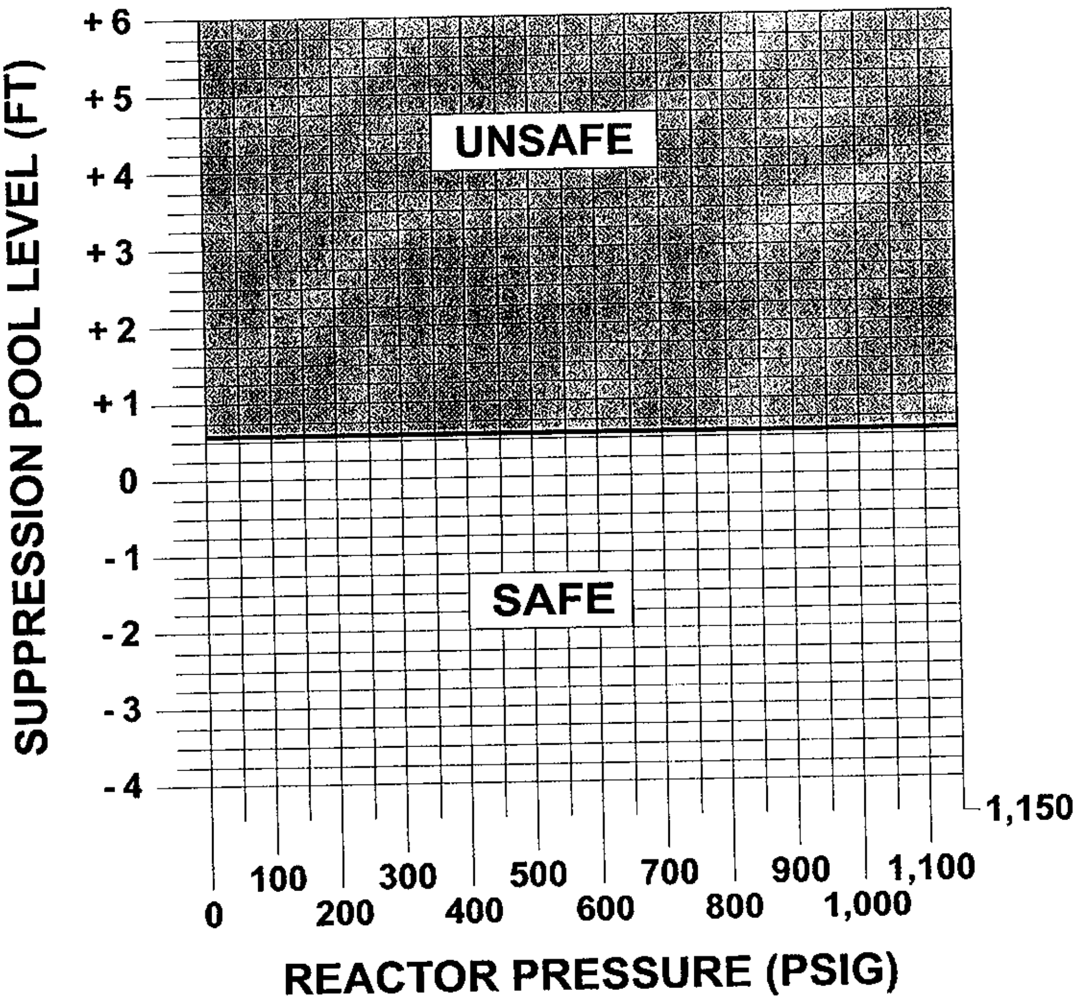
FIGURE 7A
UNIT 2 PRESSURE SUPPRESSION PRESSURE



ATTACHMENT 5 (Cont'd)

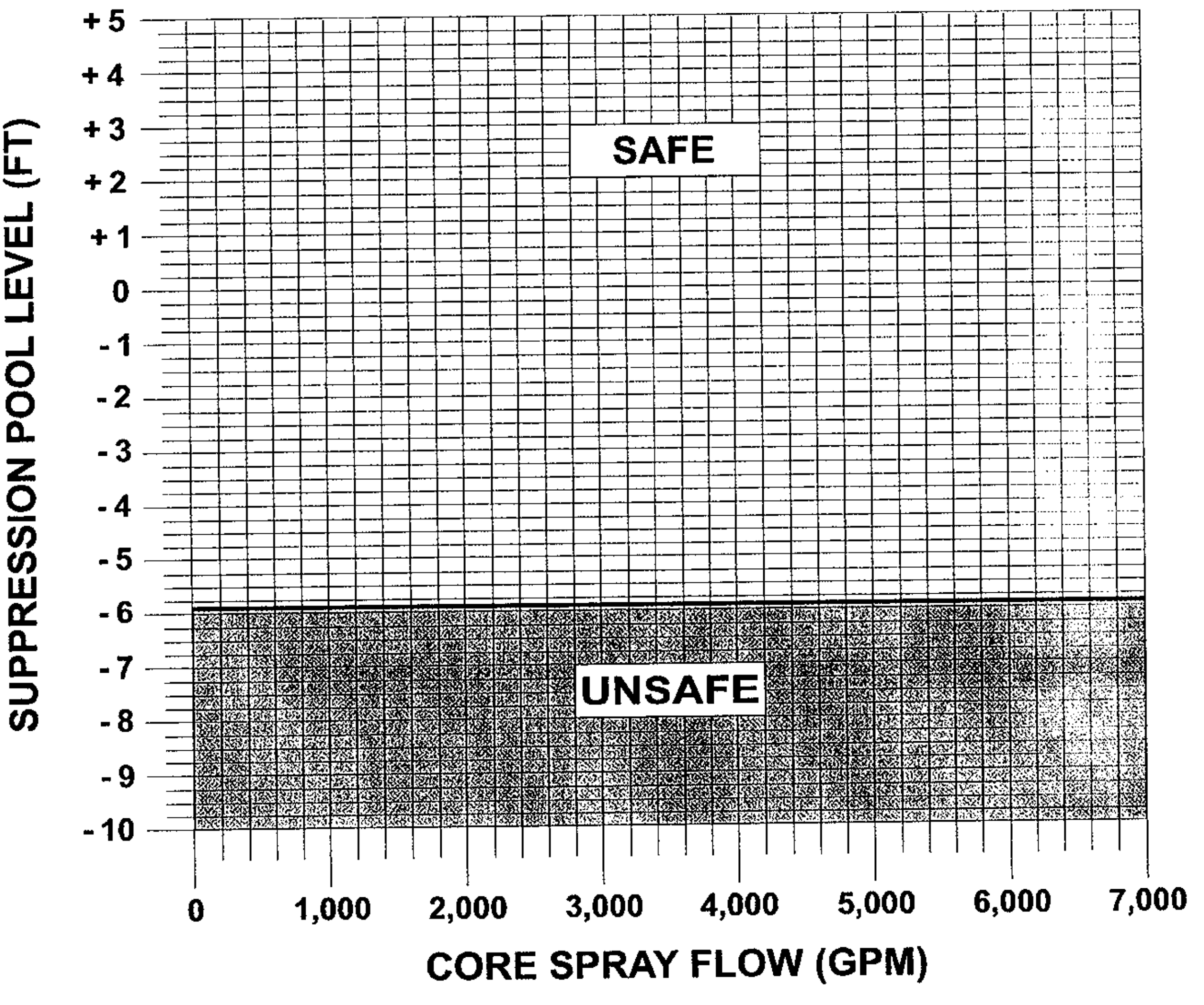
FIGURE 8

SRV TAILPIPE LEVEL LIMIT



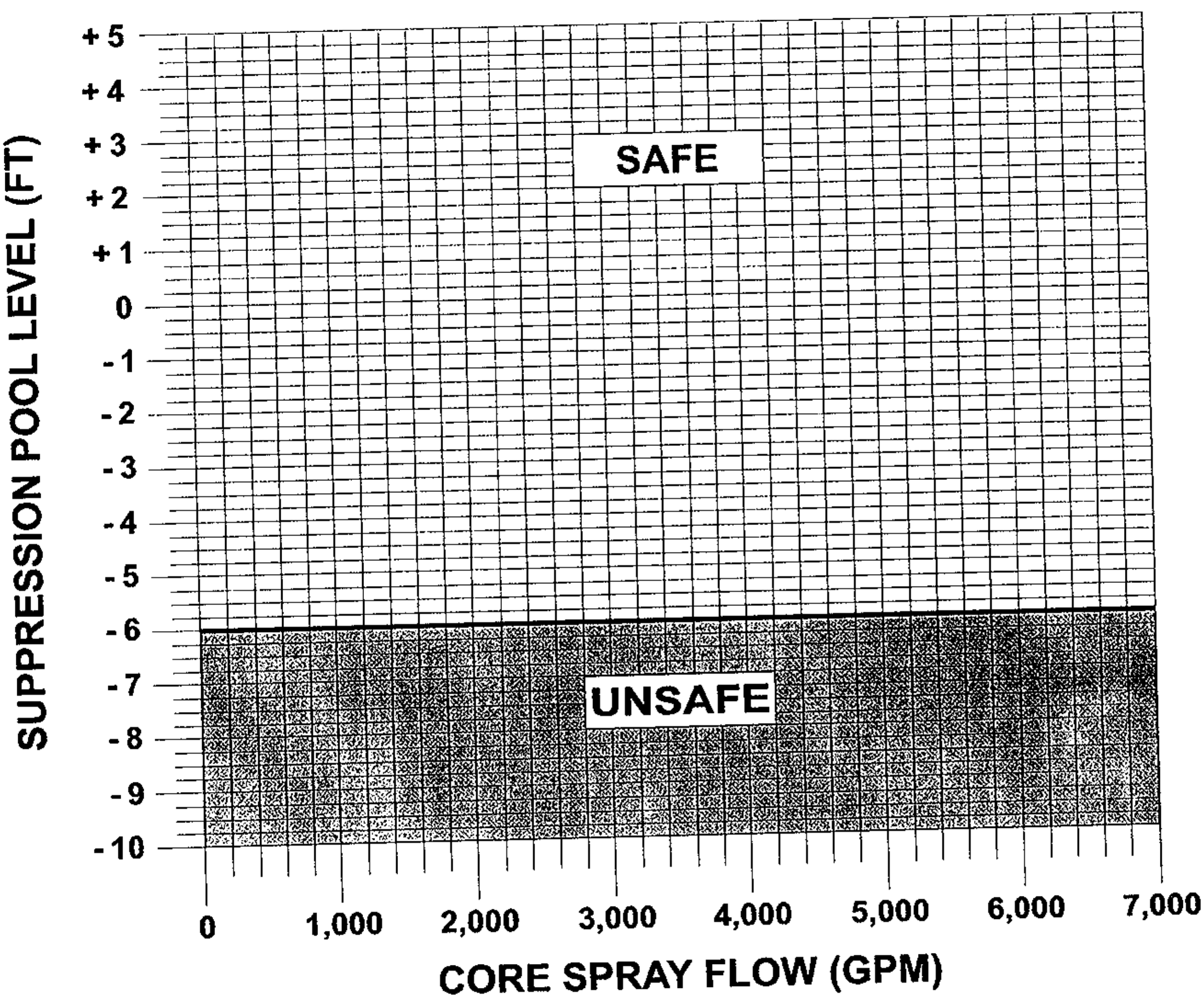
ATTACHMENT 5 (Cont'd)

FIGURE 9
UNIT 1 CORE SPRAY VORTEX LIMIT



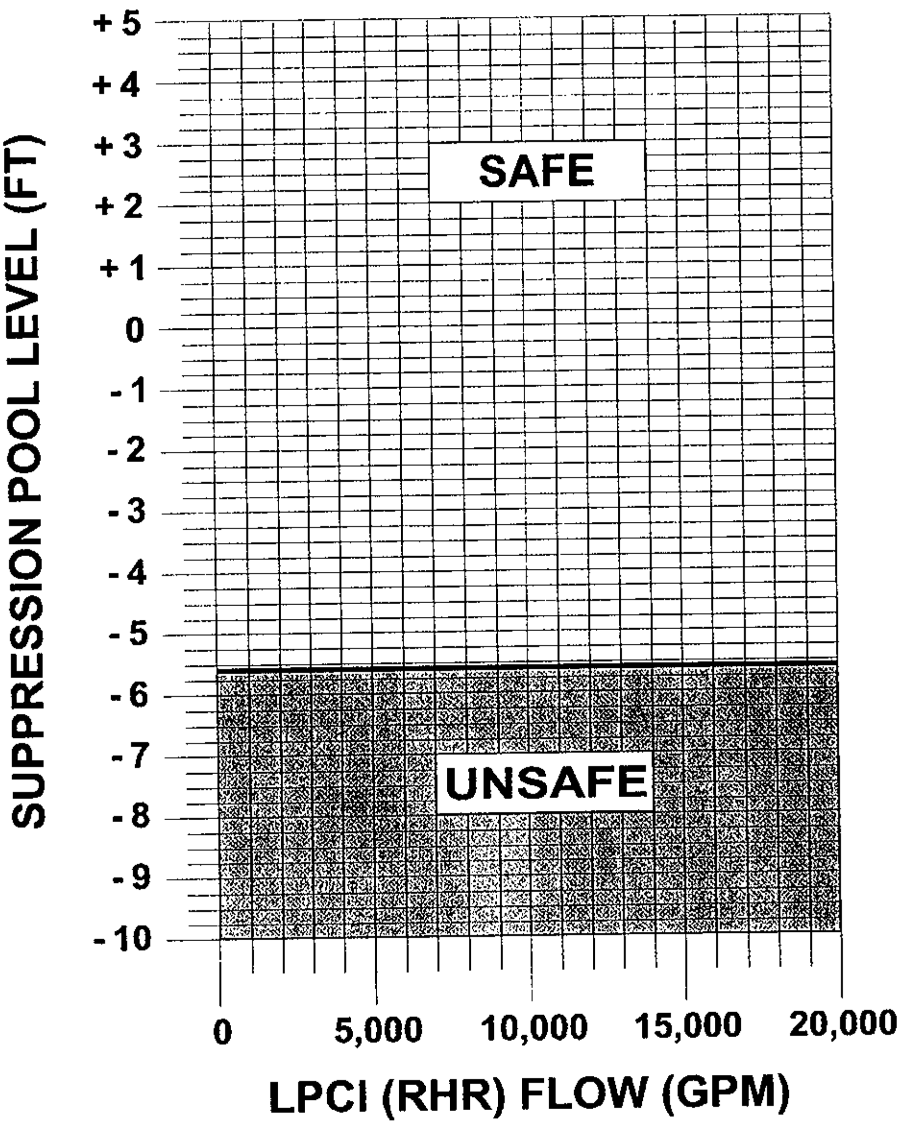
ATTACHMENT 5 (Cont'd)

FIGURE 10
UNIT 2 CORE SPRAY VORTEX LIMIT



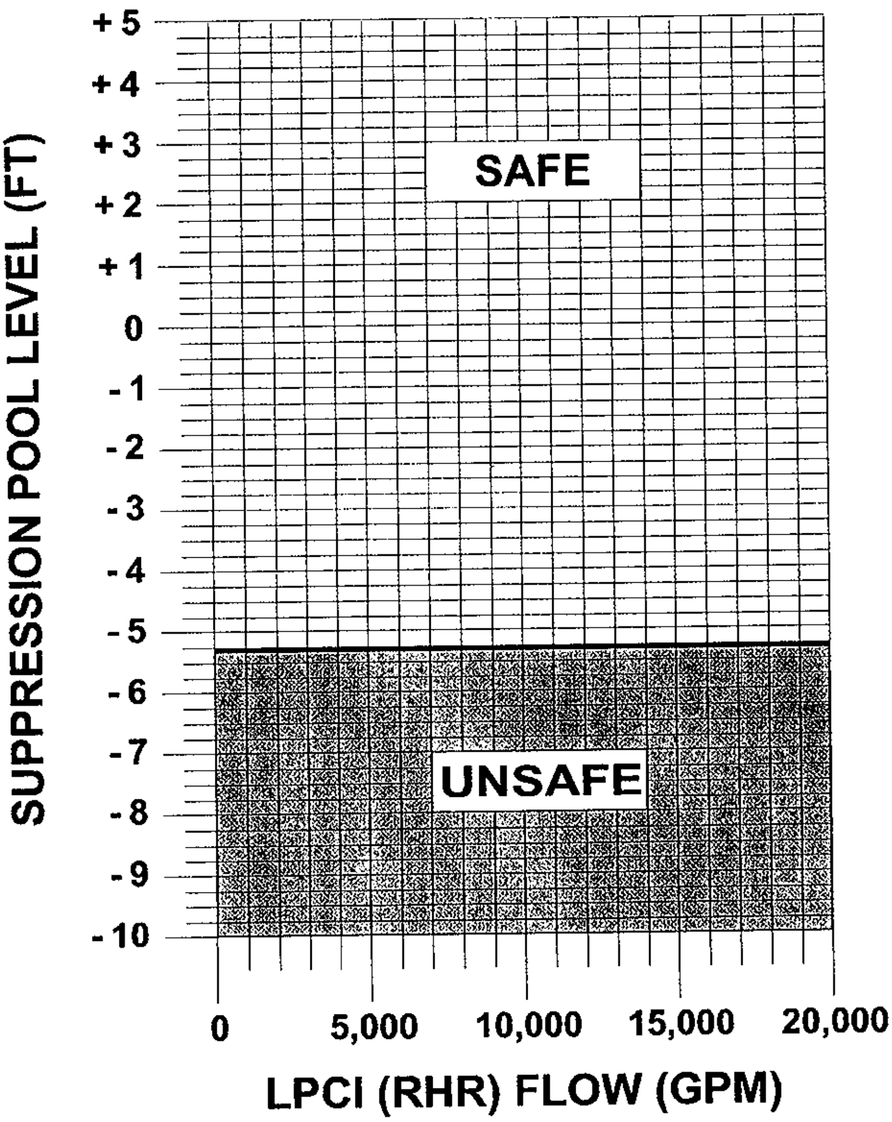
ATTACHMENT 5 (Cont'd)

FIGURE 11
UNIT 1 RHR VORTEX LIMIT



ATTACHMENT 5 (Cont'd)

FIGURE 12
UNIT 2 RHR VORTEX LIMIT



EOP-01-UG
Attachment 10
Secondary Containment Temperature And Radiation Limits

3.6 CONTAINMENT SYSTEMS

3.6.1.6 Suppression Chamber-to-Drywell Vacuum Breakers

LCO 3.6.1.6 Eight suppression chamber-to-drywell vacuum breakers shall be OPERABLE for opening.

AND

Ten suppression chamber-to-drywell vacuum breakers shall be closed, except when performing their intended function.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required suppression chamber-to-drywell vacuum breaker inoperable for opening.	A.1 Restore one vacuum breaker to OPERABLE status.	72 hours
B. One suppression chamber-to-drywell vacuum breaker not closed.	B.1 Close the open vacuum breaker.	4 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.6.1	------------------NOTE------------------ Not required to be met for vacuum breakers that are open during Surveillances. --- Verify each vacuum breaker is closed.	14 days AND Within 6 hours after any discharge of steam to the suppression chamber from any source
SR 3.6.1.6.2	Perform a functional test of each required vacuum breaker.	31 days AND Within 12 hours after any discharge of steam to the suppression chamber from any source
SR 3.6.1.6.3	Verify the full open setpoint of each required vacuum breaker is ≤ 0.5 psid.	24 months

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.6 Suppression Chamber-to-Drywell Vacuum Breakers

BASES

BACKGROUND

The function of the suppression-chamber-to-drywell vacuum breakers is to relieve vacuum in the drywell. There are 10 internal vacuum breakers located on the vent header of the vent system between the drywell and the suppression chamber, which allow flow from the suppression chamber atmosphere to the drywell when the drywell is at a negative pressure with respect to the suppression chamber. Therefore, suppression chamber-to-drywell vacuum breakers prevent an excessive negative differential pressure across the suppression chamber-drywell boundary. Each vacuum breaker is a self actuating valve, similar to a check valve, which can be remotely operated for testing purposes.

A negative differential pressure across the drywell wall is caused by depressurization of the drywell. Events that cause this depressurization are cooling cycles, inadvertent drywell spray actuation, and steam condensation from sprays or subcooled water reflood of a break in the event of a primary system rupture. Cooling cycles result in minor pressure transients in the drywell that occur slowly and are normally controlled by heating and ventilation equipment. Spray actuation or spill of subcooled water out of a break results in more significant pressure transients and becomes important in sizing the internal vacuum breakers.

In the event of a primary system rupture, steam condensation within the drywell results in the most severe pressure transient. Following a primary system rupture, the drywell atmosphere is purged into the suppression chamber free airspace, leaving the drywell full of steam. Subsequent condensation of the steam can be caused in two possible ways, namely, Emergency Core Cooling Systems flow from a recirculation line break, or drywell spray actuation following a loss of coolant accident (LOCA). These two cases determine the maximum depressurization rate of the drywell.

In addition, the waterleg in the Mark I Vent System downcomer is controlled by the drywell-to-suppression chamber differential pressure. If the drywell pressure is less than the suppression chamber pressure, there will be an

(continued)

BASES

BACKGROUND (continued)	increase in the height of the downcomer waterleg. This will result in an increase in the water clearing inertia in the event of a postulated LOCA, resulting in an increase in the peak drywell pressure. This in turn will result in an increase in the pool swell dynamic loads. The internal vacuum breakers limit the height of the waterleg in the vent system during normal operation.
APPLICABLE SAFETY ANALYSES	Analytical methods and assumptions involving the suppression chamber-to-drywell vacuum breakers are presented in Reference 1 as part of the accident response of the primary containment systems. Internal (suppression chamber-to-drywell) and external (reactor building-to-suppression chamber) vacuum breakers are provided as part of the primary containment to limit the negative differential pressure across the drywell and suppression chamber walls that form part of the primary containment boundary. The safety analyses assume that the internal vacuum breakers are closed initially and are fully open at a differential pressure of 0.5 psid (Ref. 1). Additionally, 3 of the 10 internal vacuum breakers are assumed to fail in a closed position (Ref. 1). The results of the analyses show that the design pressure is not exceeded even under the worst case accident scenario. The vacuum breaker opening differential pressure setpoint and the requirement that 8 of 10 vacuum breakers be OPERABLE (the additional vacuum breaker is required to meet the single failure criterion) are a result of the requirement placed on the vacuum breakers to limit the vent system waterleg height. The total cross sectional area of the main vent system between the drywell and suppression chamber needed to fulfill this requirement has been established as a minimum of 51.5 times the total break area. In turn, the vacuum relief capacity between the drywell and suppression chamber should be 1/16 of the total main vent cross sectional area, with the valves set to operate at $\leq$ 0.5 psid differential pressure. Design Basis Accident (DBA) analyses assume the vacuum breakers to be closed initially and to remain closed and leak tight, until the suppression pool is at a positive pressure relative to the drywell. The suppression chamber-to-drywell vacuum breakers satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 2).

(continued)

BASES (continued)

LCO

Only 8 of the 10 vacuum breakers must be OPERABLE for opening. All suppression chamber-to-drywell vacuum breakers, however, are required to be closed (except when the vacuum breakers are performing their intended design function). The vacuum breaker OPERABILITY requirement provides assurance that the drywell-to-suppression chamber negative differential pressure remains below the design value. The requirement that the vacuum breakers be closed ensures that there is no excessive bypass leakage should a LOCA occur.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could result in excessive negative differential pressure across the drywell wall, caused by the rapid depressurization of the drywell. The event that results in the limiting rapid depressurization of the drywell is the primary system rupture that purges the drywell atmosphere and fills the drywell free airspace with steam. Subsequent condensation of the steam would result in depressurization of the drywell. The limiting pressure and temperature of the primary system prior to a DBA occur in MODES 1, 2, and 3.

In MODES 4 and 5, the probability and consequences of these events are reduced by the pressure and temperature limitations in these MODES; therefore, maintaining suppression chamber-to-drywell vacuum breakers OPERABLE is not required in MODE 4 or 5.

ACTIONS

A.1

With one of the required vacuum breakers inoperable for opening (e.g., the vacuum breaker is not open and may be stuck closed or not within its opening setpoint limit, so that it would not function as designed during an event that depressurized the drywell), the remaining seven OPERABLE vacuum breakers are capable of providing the vacuum relief function. However, overall system reliability is reduced because a single failure in one of the remaining vacuum breakers could result in an excessive suppression chamber-to-drywell differential pressure during a DBA. Therefore, with one of the eight required vacuum breakers inoperable, 72 hours is allowed to restore at least one of the inoperable vacuum breakers to OPERABLE status so that

(continued)

BASES

ACTIONS

A.1 (continued)

plant conditions are consistent with those assumed for the design basis analysis. The 72 hour Completion Time is considered acceptable due to the low probability of an event in which the remaining vacuum breaker capability would not be adequate.

B.1

With one vacuum breaker not closed, communication between the drywell and suppression chamber airspace could occur, and, as a result, there is the potential for primary containment overpressurization due to this bypass leakage if a LOCA were to occur. Therefore, the open vacuum breaker must be closed. A short time is allowed to close the vacuum breaker due to the low probability of an event that would pressurize primary containment. If vacuum breaker position indication is not available, an alternate method of verifying that the vacuum breakers are closed is to verify that the differential pressure between the suppression chamber and drywell is maintained > 0.5 times the initial differential pressure for 1 hour without nitrogen makeup. The 4 hour Completion Time is considered adequate to perform this test.

C.1 and C.2

If any Required Action and associated Completion Time can not be met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.6.1

Each vacuum breaker is verified closed (except when the vacuum breaker is performing its intended design function) to ensure that this potential large bypass leakage path is not present. This Surveillance is performed by observing the vacuum breaker position indication or by verifying that

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.6.1 (continued)

the differential pressure between the suppression chamber and drywell is maintained > 0.5 times the initial differential pressure for 1 hour without nitrogen makeup. The 14 day Frequency is based on engineering judgment, is considered adequate in view of other indications of vacuum breaker status available to operations personnel and procedural controls to ensure the drywell is normally maintained at a higher pressure than the suppression chamber, and has been shown to be acceptable through operating experience. This verification is also required within 6 hours after any discharge of steam to the suppression chamber from any source.

A Note is added to this SR which allows suppression chamber-to-drywell vacuum breakers opened in conjunction with the performance of a Surveillance to not be considered as failing this SR. These periods of opening vacuum breakers are controlled by plant procedures and do not represent inoperable vacuum breakers.

SR 3.6.1.6.2

Each required vacuum breaker must be cycled to ensure that it opens adequately to perform its design function and returns to the fully closed position. This is accomplished by verifying each required vacuum breaker operates through at least one complete cycle of full travel. This SR ensures that the safety analysis assumptions are valid. The 31 day Frequency of this SR was developed, based on Inservice Testing Program requirements to perform valve testing at least once every 92 days. A 31 day Frequency was chosen to provide additional assurance that the vacuum breakers are OPERABLE, since they are located in a harsh environment (the suppression chamber airspace). In addition, this functional test is required within 12 hours after a discharge of steam to the suppression chamber from any source.

SR 3.6.1.6.3

Verification of the vacuum breaker opening setpoint is necessary to ensure that the safety analysis assumption regarding vacuum breaker full open differential pressure of 0.5 psid is valid. The 24 month Frequency is based on the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.6.3 (continued)

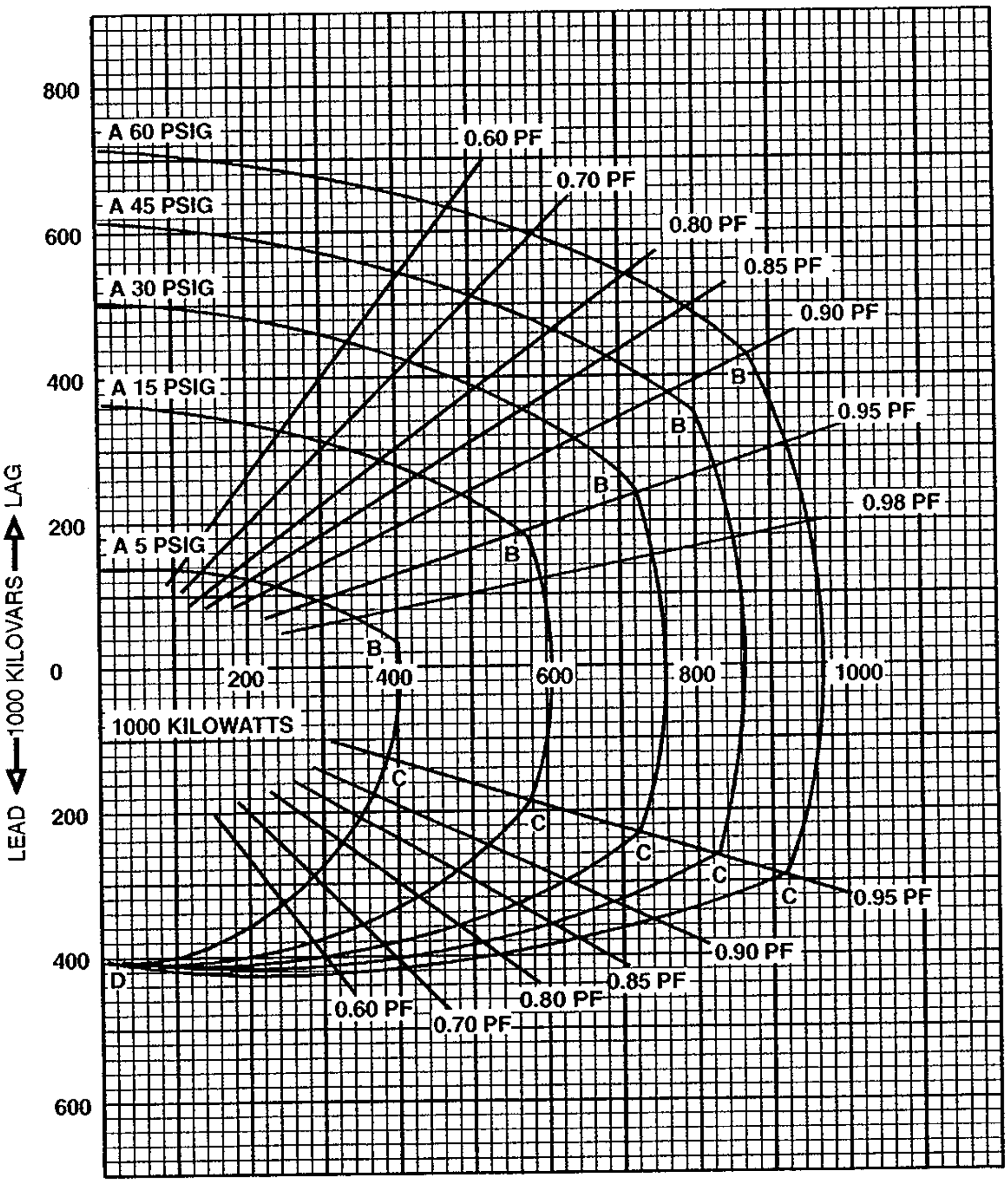
need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 24 month Frequency has been demonstrated to be acceptable, based on operating experience, and is further justified because of other surveillances performed more frequently that convey the proper functioning status of each vacuum breaker.

REFERENCES

1. UFSAR, Section 6.2.
2. 10 CFR 50.36(c)(2)(ii).

FIGURE 1
Page 1 of 1
Estimated Capability Curves

ATB 4 POLE, 963,000 KVA, 1800 RPM, 24,000 VOLTS
0.90 P.F. 0.58 SCR, 60 PSIG HYDROGEN PRESSURE, 500 VOLTS EXCITATION



CURVE AB LIMITED BY FIELD HEATING
CURVE BC LIMITED BY ARMATURE HEATING
CURVE CD LIMITED BY ARMATURE CORE END HEATING

GENERATOR AUTO TRIP TO MANUAL

AUTO ACTIONS

1. If the alarm was caused by an exciter field overcurrent, the generator backup lockup is energized (refer to APP UA-13 1-3, GEN-XFMR BACKUP L/O UNIT TRIP).
2. If shift to manual was caused by a loss of control power, the regulator will shift back to AUTO and reflash the field when control power is restored.
3. If shift to manual was caused by overexcitation and the excitation has not returned to less than or equal to 100% in 5 seconds, the generator backup lockup is energized (refer to APP UA-13 1-3, GEN-XFMR BACKUP L/O UNIT TRIP).
4. If shift to manual is due to volts/hertz being excessive, the following actions will occur:
   a. If generator is tied to grid, no actions result.
   b. If generator is not tied to grid, the following actions will occur:
      (1) Use of voltage regulator will be blocked.
      (2) Regulator will run back to no load.
      (3) If excessive volts/hertz signal is not cleared in 60 seconds, the exciter field breaker will trip.

CAUSES

1. Exciter field overcurrent.
2. Generator field overexcitation.
3. Excessive volts/hertz in exciter.
4. Loss of DC control power.
5. Circuit malfunction.

OBSERVATIONS

1. GEN-XFMR BACKUP L/O UNIT TRIP (UA-13 1-3) alarm.
2. GENERATOR FIELD OVEREXCITATION (UA-13 2-4) alarm.
3. GENERATOR EXC FIELD OVERCURRENT (UA-13 3-4) alarm.
4. Regulator shifts to manual.

ACTIONS

1. Notify Load Dispatcher of the problem.
2. If the voltage regulator mode swaps to manual, place the voltage regulator selector switch in MANUAL and perform the following:
   a. If cause of alarm was momentary, try to determine cause of alarm and verify system parameters have returned to normal.
   b. When cause for alarm is no longer a concern, return the voltage regulator selector switch to auto.
3. If the generator backup lockout is energized, refer to APP UA-13 1-3, GEN-XFMR BACKUP L/O UNIT TRIP.
4. If Circuit Breaker 2 (control power) in 125V DC Distribution Panel 10A is tripped or off, reset and close the breaker.
5. If Circuit Breaker 2 in 125V DC Distribution Panel 10A trips again, ensure that a WR/WO is prepared.

DEVICE/SETPOINTS

Voltage regulator control switch	AUTO
AND	
Generator Exciter Field Overcurrent Relay 76/50	400 amps instantaneous overcurrent or 180 amps @ 60 seconds
Overexcitation Relay J1K	105%
Volts/Hertz Relay 43T	Energized

POSSIBLE PLANT EFFECTS

1. Loss of unit generator.
2. If generator trips, possible reactor Scram.

REFERENCES

1. 9527-LL-9351 - 34
2. APP UA-13 1-3, GEN XFMR BACKUP L/O UNIT TRIP
3. GEK-33798 Vol. II, Generator Section

VOLT BALANCE RELAY A OPERATION

AUTO ACTIONS

1. Transfers excitation to manual.
2. Prevents generator loss of field relay (40-1) from actuating.
3. Prevents generator voltage restrained time overcurrent relay (51V-1) from actuating.
4. Prevents generator directional distant relay (21G-1) from actuating.

CAUSE

1. Decreased voltage balance (80% reduction).
2. Circuit malfunction.

OBSERVATIONS

NONE

ACTIONS

1. As necessary, adjust generator excitation to maintain voltage.
2. If a circuit or equipment malfunction is suspected, ensure that a WR/JO is prepared.

DEVICE/SETPOINTS

Generator Voltage Balance Relay 60-1 Right	80% balanced reduction

POSSIBLE PLANT EFFECTS

1. Loss of automatic voltage control.
2. Loss of some generator protective relaying.

REFERENCES

9527-LL-9361 - 15

GEN LOSS OF EXC

AUTO ACTIONS

1. Energizes the generator primary lockout relays (refer to APP UA-13 1-1, GEN-XFMR PRIMARY L/O UNIT TRIP).
2. Energizes the generator breaker failure lockout relays if the generator failed to trip on the generator primary lockout relays, and if an instantaneous phase or ground overcurrent condition exists on the breaker.

CAUSES

1. Loss of generator excitation.
2. Circuit malfunction.

OBSERVATIONS

1. GEN-XFMR PRIMARY L/O UNIT TRIP (UA-13 1-1) alarms.

ACTIONS

1. Refer to APP UA-13 1-1, GEN-XFMR PRIMARY L/O UNIT TRIP.

DEVICE/SETPOINTS

Loss of Field Relay 40	20% restraint

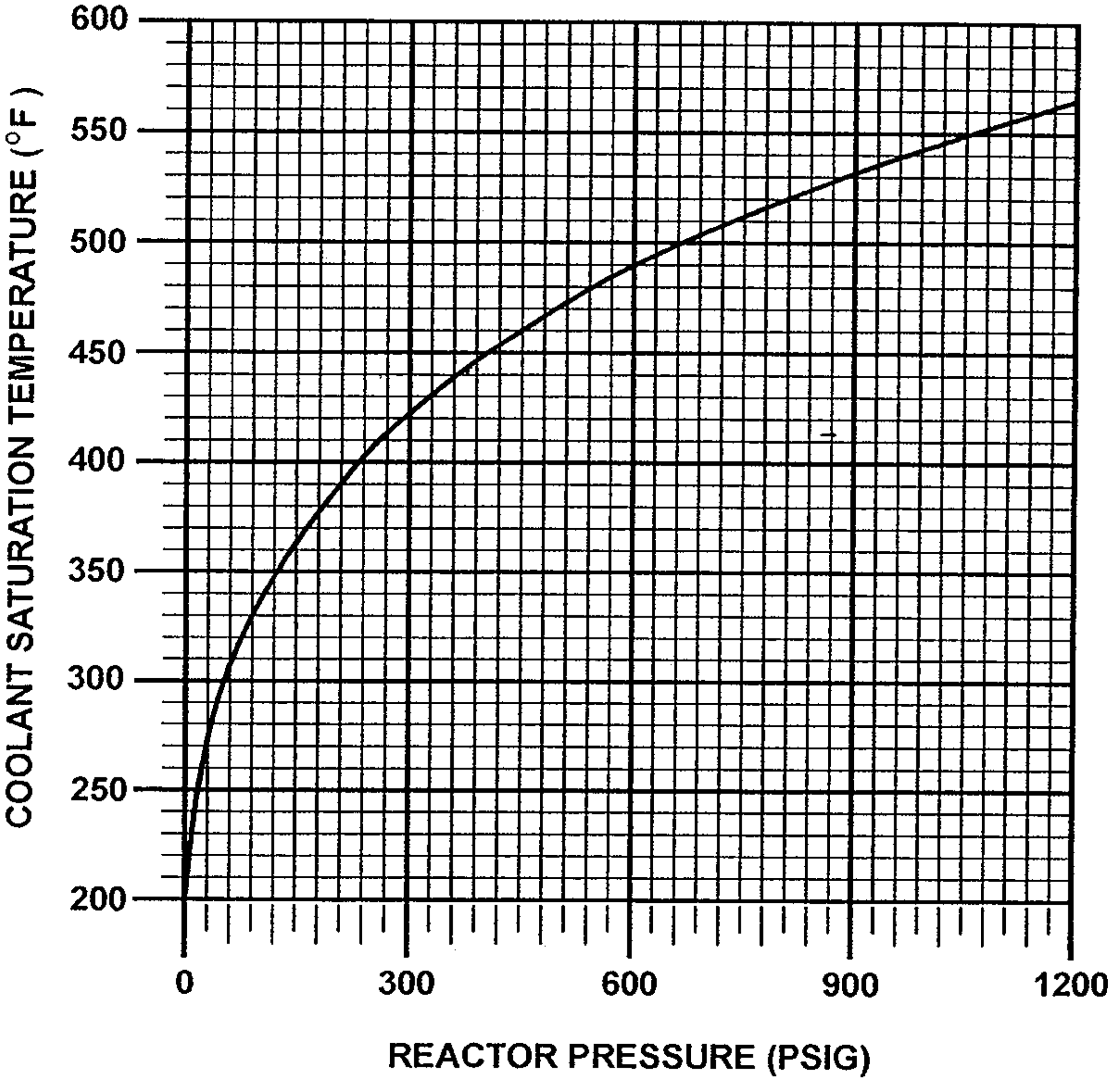
POSSIBLE PLANT EFFECTS

1. Loss of unit generator.

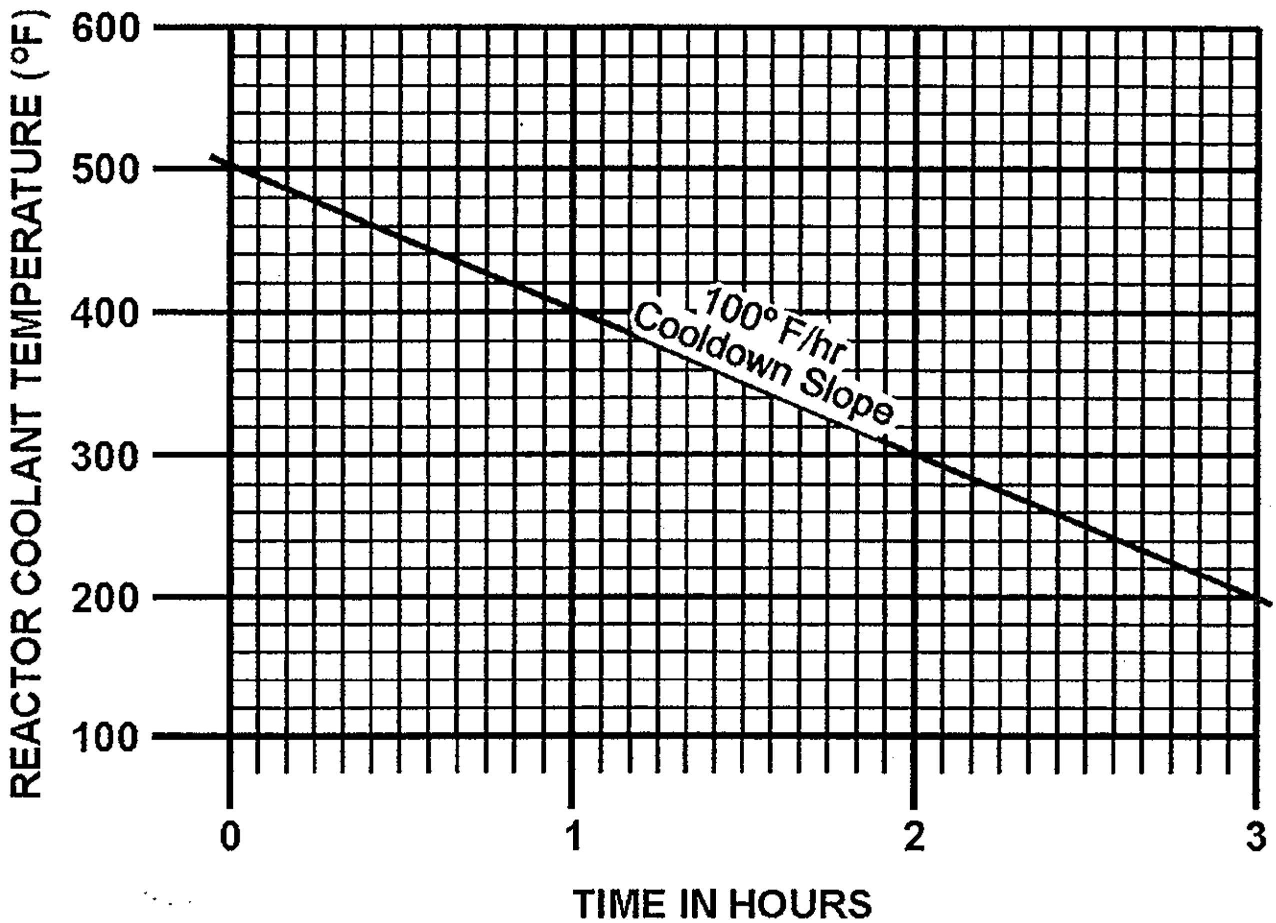
REFERENCES

1. 9527-LL-9351 - 28
2. APP UA-13 1-1, GEN-XFMR PRIMARY L/O UNIT TRIP

Reactor Pressure vs Saturation Temperature



Reactor Cooldown Plot



REACTOR VESS HI PRESS

AUTOMATIC ACTIONS

NONE

CAUSE

1. MSIV closure.
2. MSIV failure (disk/stem separation)
3. EHC System malfunction.
4. Pressure setpoint set too high.
5. Circuit malfunction.

OBSERVATIONS

1. MSIVs indicating closed.
2. One of the steam line flow indicators indicating no flow with associated MSIVs indicating open indicates a disk/stem separation.
3. Turbine control valves, stop valves, or bypass valves closing indicates an EHC System malfunction.
4. Pressure setpoint set greater than 945 psig.
5. REACTOR VESSEL HI PRESS alarm on with reactor pressure less than 1050 psig indicates a defective trip unit.

ACTIONS

1. If a reactor Scram occurs, refer to EOP-01-RSP.
2. For a disk/stem separation:
   a. Close the MSIVs associated with blocked steam line.
   b. Notify the Reactor Engineer that new core analysis is needed.
3. If pressure setpoint is set too high, reduce reactor pressure to 1030 psig.
4. If a circuit malfunction is suspected, ensure that a WR/JO is prepared.

DEVICE/SETPOINTS

Pressure Trip Unit B21-PTS-N023A-2	1050 psig
Pressure Trip Unit B21-PTS-N023B-2	1050 psig
Pressure Trip Unit B21-PTS-N023C-2	1050 psig
Pressure Trip Unit B21-PTS-N023D-2	1050 psig

POSSIBLE PLANT EFFECTS

1. Reactor Scram if pressure increases to 1060 psig.

REFERENCES

1. LL-9364 -79
2. EOP-01-RSP, Reactor Scram Procedure

ATTACHMENT 10
SECONDARY CONTAINMENT TEMPERATURE AND RADIATION LIMITS

FIGURE 22
SECONDARY CONTAINMENT AREA TEMPERATURE

TABLE 1
AREA TEMPERATURE LIMITS

PLANT AREA	PLANT LOCATION DESCRIPTION	STEAM LEAK DETECTION CHANNEL/LOCATION	INSTRUMENT NUMBER/ WINDOW	MAX NORM OPERATING VALUE (°F) (NOTE 1)	MAX SAFE OPERATING VALUE (°F)	AUTO GROUP ISOL
N CORE SPRAY	N CORE SPRAY ROOM	PANEL XU-3	VA-TI-1603	120	175	N/A
S CORE SPRAY	S CORE SPRAY ROOM	PANEL XU-3	VA-TI-1604	120	175	N/A
RWCU	RWCU PUMP ROOM A	B21-XY-5949A B21-XY-5949B CH. A1-1	G31-TE-N016A G31-TE-N016B	140	225	3
	RWCU PUMP ROOM B	B21-XY-5949A B21-XY-5949B CH. A2-1	G31-TE-N016C G31-TE-N016D			
	RWCU HX ROOM	B21-XY-5949A B21-XY-5949B CH. A3-1	G31-TE-N016E G31-TE-N016F			
N RHR	N RHR EQUIP ROOM	B21-XY-5948A CH. A5-4 PANEL XU-3	E11-TE-N009A VA-TI-1601	175	295	N/A
S RHR	S RHR EQUIP ROOM	B21-XY-5948B CH. A5-4 PANEL XU-3	E11-TE-N009B VA-TI-1602	175	295	N/A
	RCIC EQUIP ROOM	B21-XY-5949A B21-XY-5949B CH. A1-3	E51-TE-N023A E51-TE-N023B	165	295	5
HPCI	HPCI EQUIP ROOM	B21-XY-5948A B21-XY-5948B CH. A1-1	E41-TE-N030A E41-TE-N030B	165		4
STEAM TUNNEL	RCIC STM TUNNEL	B21-XY-5949A B21-XY-5949B CH. A3-3	E51-TE-N025A E51-TE-N025B	190	295	5
	HPCI STM TUNNEL	B21-XY-5948A B21-XY-5948B CH. A5-1	E51-TE-N025C E51-TE-N025D	190	295	4
20 FT	20 FT NORTH	B21-XY-5948A CH. A1-4	B21-TE-5761A	140	200	N/A
	20 FT SOUTH	B21-XY-5948B CH. A1-4	B21-TE-5763B			
50 FT	50 FT NW	B21-XY-5948A CH. A2-4	B21-TE-5762A	140	200	N/A
	50 FT SE	B21-XY-5948B CH. A2-4	B21-TE-5764B			
REACTOR BLDG	MULTIPLE AREAS	ANNUNCIATOR PANEL A-02	WINDOW 5-7	ALARM SETPOINT	N/A	3,4, AND/OR 5
REACTOR BLDG	MSIV PIT	ANNUNCIATOR PANEL A-06	WINDOW 6-7	ALARM SETPOINT	N/A	1

NOTE 1 MAX NORM OPERATING VALUE IS THE ANNUNCIATOR/GROUP ISOLATION SETPOINT WHERE APPLICABLE

ATTACHMENT 10 (Cont'd)

FIGURE 23
SECONDARY CONTAINMENT AREA DIFFERENTIAL TEMPERATURE

TABLE 2
AREA DIFFERENTIAL TEMPERATURE LIMITS

PLANT AREA	PLANT LOCATION DESCRIPTION	STEAM LEAK DETECTION CHANNEL	MAX NORM OPERATING VALUE (°F) (NOTE 1)	AUTO GROUP ISOL
RWCU	RWCU PUMP ROOM A	B21-XY-5949A B21-XY-5949B CH. A4-1	47	3
	RWCU PUMP ROOM B	B21-XY-5949A B21-XY-5949B CH. A5-1		
	RWCU HX ROOM	B21-XY-5949A B21-XY-5949B CH. A6-1		
N RHR	N RHR EQUIP ROOM	B21-XY-5948A CH. A6-4	50	N/A
S RHR	S RHR EQUIP ROOM	B21-XY-5948B CH. A6-4	50	N/A
	RCIC EQUIP ROOM	B21-XY-5949A B21-XY-5949B CH. A2-3	47	5
HPCI	HPCI EQUIP ROOM	B21-XY-5948A B21-XY-5948B CH. A3-1	47	N/A
STEAM TUNNEL	RCIC STM TUNNEL	B21-XY-5949A B21-XY-5949B CH. A4-3	47	5
	HPCI STM TUNNEL	B21-XY-5948A B21-XY-5948B CH. A6-1	47	4
REACTOR BLDG	MULTIPLE AREAS	ANNUNCIATOR A-02 6-7	ALARM SETPOINT	3, 4, AND/OR 5

NOTE 1: MAX NORM OPERATING VALUE IS THE ANNUNCIATOR/GROUP ISOLATION SETPOINT WHERE APPLICABLE

ATTACHMENT 10 (Cont'd)

FIGURE 24
SECONDARY CONTAINMENT AREA RADIATION

TABLE 3
AREA RADIATION LIMITS

PLANT AREA	PLANT LOCATION DESCRIPTION	ARM CHANNEL	MAX NORM OPERATING VALUE (mR/HR)	MAX SAFE OPERATING VALUE (mR/HR)
N CORE SPRAY	N CORE SPRAY ROOM	15	200	* 7000
S CORE SPRAY	S CORE SPRAY ROOM	16	200	* 7000
N RHR	N RHR ROOM	17	200	* 7000
S RHR	S RHR ROOM	18	200	* 3000
HPCI	HPCI ROOM	N/A	N/A	* 3000
RX BLDG 20 FT ELEV	N ACROSS FROM TIP ROOM	19	80	* 2000
	DRYWELL ENTRANCE	20		
	DECON ROOM	22		
	RAILROAD DOORS	23		
RX BLDG 50 FT ELEV	SAMPLE STATION	24	80	* 2000
	RX BLDG AIR LOCK	25		
RX BLDG 117 FT ELEV	N OF FUEL STORAGE POOL	27	80	* 7000
	BETWEEN RX & FUEL POOL	28	1000	7000
	CASK WASH AREA	29	90	* 7000
RX BLDG 80 FT ELEV	SPENT FUEL COOLING SYSTEM	30	90	* 3000

* CONTACT E&RC TO DETERMINE IF MAX SAFE OPERATING VALUE IS EXCEEDED

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3.	A.1	Restore secondary containment to OPERABLE status.	8 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	12 hours
	AND		
	B.2	Be in MODE 4.	36 hours
C. Secondary containment inoperable during movement of recently irradiated fuel assemblies in the secondary containment, or during OPDRVs.	C.1	--------NOTE--------- LCO 3.0.3 is not applicable. --------------------- Suspend movement of recently irradiated fuel assemblies in the secondary containment.	Immediately
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify all secondary containment equipment hatches are closed and sealed.	24 months
SR 3.6.4.1.2	Verify one secondary containment access door is closed in each access opening.	24 months
SR 3.6.4.1.3	Verify each SGT subsystem can maintain $\geq$ 0.25 inch of vacuum water gauge in the secondary containment for 1 hour at a flow rate $\leq$ 3000 cfm.	24 months on a STAGGERED TEST BASIS

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.1 Secondary Containment

BASES

BACKGROUND	The function of the secondary containment is to contain and hold up fission products that may leak from primary containment following a Design Basis Accident (DBA). In conjunction with operation of the Standby Gas Treatment (SGT) System and closure of certain valves whose lines penetrate the secondary containment, the secondary containment is designed to reduce the activity level of the fission products prior to release to the environment and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside primary containment. The secondary containment is a structure that completely encloses the primary containment and those components that may be postulated to contain primary system fluid. This structure forms a control volume that serves to hold up the fission products. It is possible for the pressure in the control volume to rise relative to the environmental pressure. To prevent ground level exfiltration while allowing the secondary containment to be designed as a conventional structure, the secondary containment requires support systems to maintain the control volume pressure at less than the external pressure. Requirements for these systems are specified separately in LCO 3.6.4.2, "Secondary Containment Isolation Dampers (SCIDs)," and LCO 3.6.4.3, "Standby Gas Treatment (SGT) System."
APPLICABLE SAFETY ANALYSES	There are two principal accidents for which credit is taken for secondary containment OPERABILITY. These are a loss of coolant accident (LOCA) (Refs. 1 and 2) and a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours) inside secondary containment. The secondary containment performs no active function in response to each of these limiting events; however, its leak tightness is required to ensure that fission products entrapped within the secondary containment structure will be treated by the SGT System prior to discharge to the environment.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Secondary containment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

LCO

An OPERABLE secondary containment provides a control volume into which fission products that leak from primary containment, or are released from the reactor coolant pressure boundary components or irradiated fuel assemblies located in secondary containment, can be processed prior to release to the environment. For the secondary containment to be considered OPERABLE, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained, at least one door in each access to the Reactor Building must be closed, and the sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) must be OPERABLE.

APPLICABILITY

In MODES 1, 2, and 3, a LOCA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, secondary containment OPERABILITY is required during the same operating conditions that require primary containment OPERABILITY.

In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining secondary containment OPERABLE is not required in MODE 4 or 5 to ensure a control volume, except for other situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs) or during movement of recently irradiated fuel assemblies in the secondary containment. Due to radioactive decay, secondary containment is only required to be OPERABLE during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

ACTIONS

A.1

If secondary containment is inoperable, it must be restored to OPERABLE status within 8 hours. The 8 hour Completion Time provides a period of time to correct the problem that is commensurate with the importance of maintaining secondary

(continued)

BASES

ACTIONS

A.1 (continued)

containment during MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring secondary containment OPERABILITY) occurring during periods where secondary containment is inoperable is minimal.

B.1 and B.2

If secondary containment cannot be restored to OPERABLE status within the required Completion Time, 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 and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

Movement of irradiated fuel assemblies in the secondary containment and OPDRVs can be postulated to cause significant fission product release to the secondary containment. In such cases, the secondary containment is the only barrier to release of fission products to the environment. Therefore, movement of recently irradiated fuel assemblies must be immediately suspended if the secondary containment is operable. Suspension of this activity shall not preclude completing an action that involves moving a component to a safe position. Also, action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since recently irradiated fuel assembly movement can occur in MODE 1, 2, or 3, Required Action C.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving recently irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

of recently irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.1 and SR 3.6.4.1.2

Verifying that secondary containment equipment hatches and one secondary containment access door in each access opening are closed ensures that the infiltration of outside air of such magnitude as to prevent maintaining the desired negative pressure does not occur. Verifying that all such openings are closed provides adequate assurance that exfiltration from the secondary containment will not occur. In this application, the term "sealed" has no connotation of leak tightness. Maintaining secondary containment OPERABILITY requires verifying one door in each access opening is closed. The 24 month Frequency for these SRs has been shown to be adequate, based on operating experience, and is considered adequate in view of other indications of door and hatch status that are available to the operator.

SR 3.6.4.1.3

The SGT System exhausts the secondary containment atmosphere to the environment through appropriate treatment equipment. To ensure that fission products are treated, SR 3.6.4.1.3 verifies that the SGT System will establish and maintain a negative pressure in the secondary containment. This is confirmed by demonstrating that one SGT subsystem can maintain ≥ 0.25 inches of vacuum water gauge for 1 hour at a flow rate ≤ 3000 cfm. The 1 hour test period allows secondary containment to be in thermal equilibrium at steady state conditions. Therefore, this test is used to ensure secondary containment boundary integrity. Since this SR is a secondary containment test, it need not be performed with each SGT subsystem. The SGT subsystems are tested on a STAGGERED TEST BASIS, however, to ensure that in addition to the requirements of LCO 3.6.4.3, either SGT subsystem will perform this test. Operating experience has demonstrated these components will usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES (continued)

REFERENCES

1. NEDC-32466P, Power Uprate Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2, September 1995.

2. UFSAR, Section 15.6.4.

3. Not used.

4. 10 CFR 50.36(c)(2)(ii).

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3.	A.1	Restore secondary containment to OPERABLE status.	8 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	12 hours
	AND		
	B.2	Be in MODE 4.	36 hours
C. Secondary containment inoperable during movement of recently irradiated fuel assemblies in the secondary containment, or during OPDRVs.	C.1	--------NOTE--------- LCO 3.0.3 is not applicable. ---------------------	
		Suspend movement of recently irradiated fuel assemblies in the secondary containment.	Immediately
	AND		

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify all secondary containment equipment hatches are closed and sealed.	24 months
SR 3.6.4.1.2	Verify one secondary containment access door is closed in each access opening.	24 months
SR 3.6.4.1.3	Verify each SGT subsystem can maintain $\geq$ 0.25 inch of vacuum water gauge in the secondary containment for 1 hour at a flow rate $\leq$ 3000 cfm.	24 months on a STAGGERED TEST BASIS

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.1 Secondary Containment

BASES

BACKGROUND

The function of the secondary containment is to contain and hold up fission products that may leak from primary containment following a Design Basis Accident (DBA). In conjunction with operation of the Standby Gas Treatment (SGT) System and closure of certain valves whose lines penetrate the secondary containment, the secondary containment is designed to reduce the activity level of the fission products prior to release to the environment and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside primary containment.

The secondary containment is a structure that completely encloses the primary containment and those components that may be postulated to contain primary system fluid. This structure forms a control volume that serves to hold up the fission products. It is possible for the pressure in the control volume to rise relative to the environmental pressure. To prevent ground level exfiltration while allowing the secondary containment to be designed as a conventional structure, the secondary containment requires support systems to maintain the control volume pressure at less than the external pressure. Requirements for these systems are specified separately in LCO 3.6.4.2, "Secondary Containment Isolation Dampers (SCIDs)," and LCO 3.6.4.3, "Standby Gas Treatment (SGT) System."

APPLICABLE SAFETY ANALYSES

There are two principal accidents for which credit is taken for secondary containment OPERABILITY. These are a loss of coolant accident (LOCA) (Refs. 1 and 2) and a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours) inside secondary containment. The secondary containment performs no active function in response to each of these limiting events; however, its leak tightness is required to ensure that fission products entrapped within the secondary containment structure will be treated by the SGT System prior to discharge to the environment.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Secondary containment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

LCO

An OPERABLE secondary containment provides a control volume into which fission products that leak from primary containment, or are released from the reactor coolant pressure boundary components or irradiated fuel assemblies located in secondary containment, can be processed prior to release to the environment. For the secondary containment to be considered OPERABLE, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained, at least one door in each access to the Reactor Building must be closed, and the sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) must be OPERABLE.

APPLICABILITY

In MODES 1, 2, and 3, a LOCA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, secondary containment OPERABILITY is required during the same operating conditions that require primary containment OPERABILITY.

In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining secondary containment OPERABLE is not required in MODE 4 or 5 to ensure a control volume, except for other situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs) or during movement of recently irradiated fuel assemblies in the secondary containment. Due to radioactive decay, secondary containment is only required to be OPERABLE during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

ACTIONS

A.1

If secondary containment is inoperable, it must be restored to OPERABLE status within 8 hours. The 8 hour Completion Time provides a period of time to correct the problem that is commensurate with the importance of maintaining secondary

(continued)

BASES

ACTIONS

A.1 (continued)

containment during MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring secondary containment OPERABILITY) occurring during periods where secondary containment is inoperable is minimal.

B.1 and B.2

If secondary containment cannot be restored to OPERABLE status within the required Completion Time, 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 and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

Movement of recently irradiated fuel assemblies in the secondary containment and OPDRVs can be postulated to cause significant fission product release to the secondary containment. In such cases, the secondary containment is the only barrier to release of fission products to the environment. Therefore, movement of recently irradiated fuel assemblies must be immediately suspended if the secondary containment is inoperable. Suspension of this activity shall not preclude completing an action that involves moving a component to a safe position. Also, action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since recently irradiated fuel assembly movement can occur in MODE 1, 2, or 3, Required Action C.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving recently irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

of recently irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.1 and SR 3.6.4.1.2

Verifying that secondary containment equipment hatches and one secondary containment access door in each access opening are closed ensures that the infiltration of outside air of such magnitude as to prevent maintaining the desired negative pressure does not occur. Verifying that all such openings are closed provides adequate assurance that exfiltration from the secondary containment will not occur. In this application, the term "sealed" has no connotation of leak tightness. Maintaining secondary containment OPERABILITY requires verifying one door in each access opening is closed. The 24 month Frequency for these SRs has been shown to be adequate, based on operating experience, and is considered adequate in view of other indications of door and hatch status that are available to the operator.

SR 3.6.4.1.3

The SGT System exhausts the secondary containment atmosphere to the environment through appropriate treatment equipment. To ensure that fission products are treated, SR 3.6.4.1.3 verifies that the SGT System will establish and maintain a negative pressure in the secondary containment. This is confirmed by demonstrating that one SGT subsystem can maintain $\geq$ 0.25 inches of vacuum water gauge for 1 hour at a flow rate $\leq$ 3000 cfm. The 1 hour test period allows secondary containment to be in thermal equilibrium at steady state conditions. Therefore, this test is used to ensure secondary containment boundary integrity. Since this SR is a secondary containment test, it need not be performed with each SGT subsystem. The SGT subsystems are tested on a STAGGERED TEST BASIS, however, to ensure that in addition to the requirements of LCO 3.6.4.3, either SGT subsystem will perform this test. Operating experience has demonstrated these components will usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES (continued)

REFERENCES

1. NEDC-32466P, Power Uprate Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2, September 1995.
2. UFSAR, Section 15.6.4.
3. Not used.
4. 10 CFR 50.36(c)(2)(ii).
5. 10 CFR 50.36(c) (2) (ii).
6. Regulatory Guide 1.52, Revision 1.

Table 1

MCPR Limits

(EOC-RPT Not Required)

<table>
<tr><th colspan="4">Steady State, Non-pressurization Transient MCPR Limits</th></tr>
<tr><td colspan="2">Fuel Type</td><td colspan="2">Exposure Range: BOC - EOC</td></tr>
<tr><td colspan="2">GE13</td><td colspan="2">1.29</td></tr>
<tr><td colspan="2">A10</td><td colspan="2">1.43</td></tr>
<tr><th colspan="4">Pressurization Transient MCPR Limits, OLMCPR (100%P): Turbine Bypass System Operable</th></tr>
<tr><th rowspan="2">MCPR Option</th><th rowspan="2">Fuel Type</th><td colspan="2">Normal and Reduced Feedwater Temperature</td></tr>
<tr><td>Exposure Range:
BOC to EOFPC-2205 MWd/MT</td><td>Exposure Range:
EOFPC-2205 MWd/MT to EOC</td></tr>
<tr><td rowspan="2">A</td><td>GE13</td><td>1.39</td><td>1.46</td></tr>
<tr><td>A10</td><td>1.55</td><td>1.62</td></tr>
<tr><td rowspan="2">B</td><td>GE13</td><td>1.34</td><td>1.38</td></tr>
<tr><td>A10</td><td>1.49</td><td>1.53</td></tr>
<tr><th colspan="4">Pressurization Transient MCPR Limits, OLMCPR (100%P): Turbine Bypass System Inoperable</th></tr>
<tr><th>MCPR Option</th><th>Fuel Type</th><td colspan="2">Normal and Reduced Feedwater Temperature
BOC to EOC</td></tr>
<tr><td rowspan="2">A</td><td>GE13</td><td colspan="2">1.48</td></tr>
<tr><td>A10</td><td colspan="2">1.65</td></tr>
<tr><td rowspan="2">B</td><td>GE13</td><td colspan="2">1.40</td></tr>
<tr><td>A10</td><td colspan="2">1.56</td></tr>
</table>

This Table is referred to by Technical Specifications 3.2.2, 3.4.1 and 3.7.6.

Figure 11

GE13 Flow-Dependent MCPR Limit, MCPR(F)

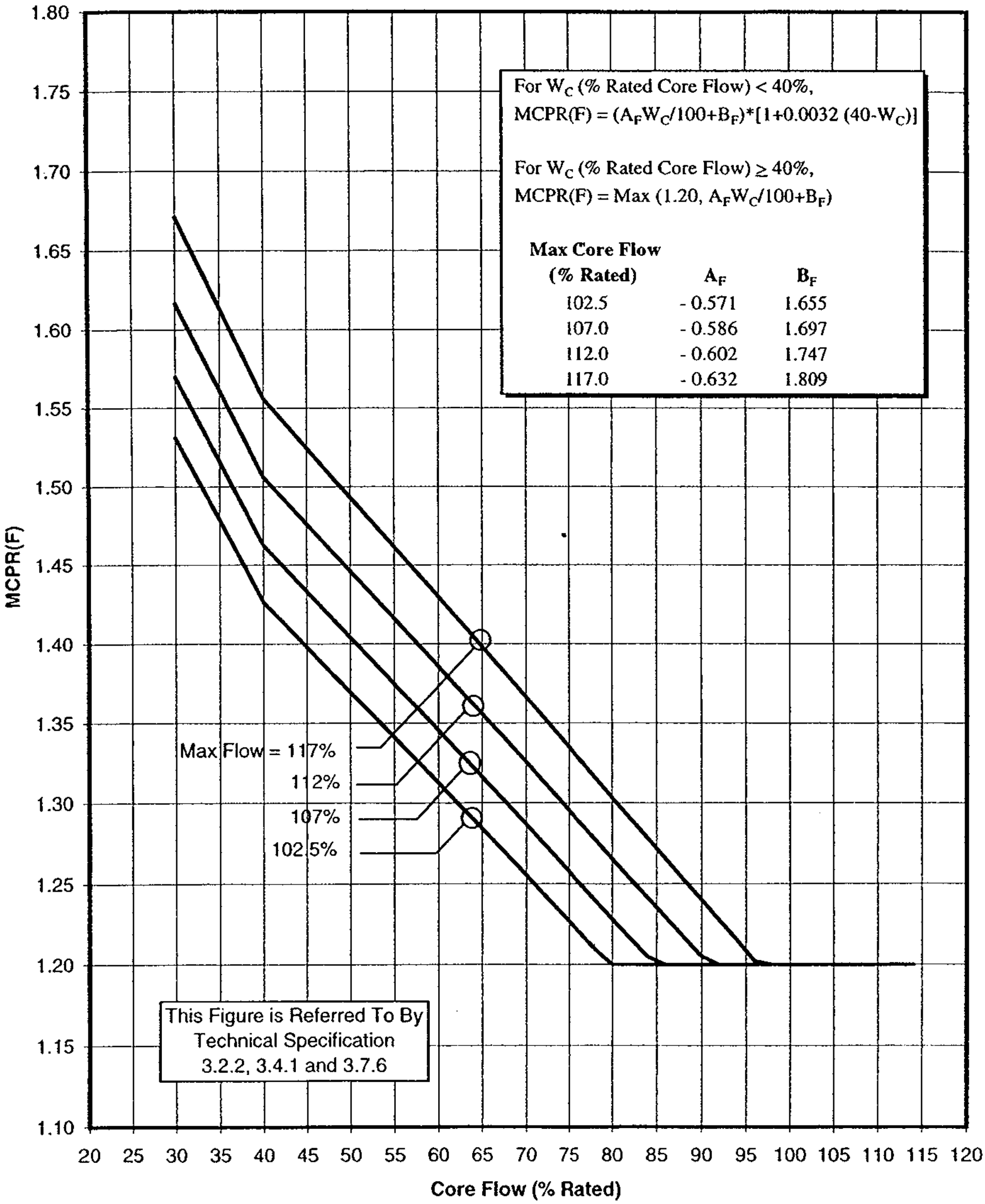
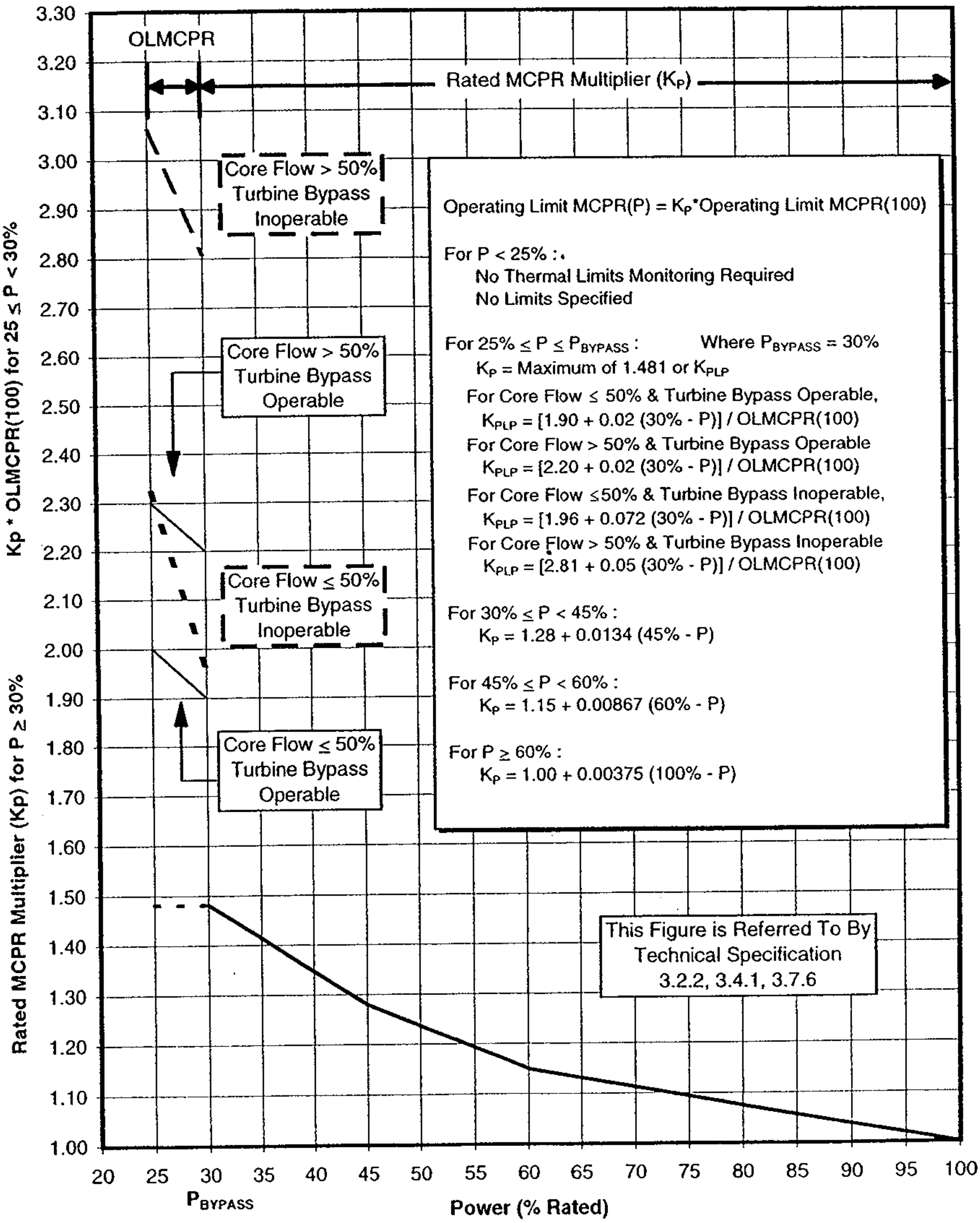


Figure 12

Power - Dependent MCPR Limit, MCPR (P)



3.6 CONTAINMENT SYSTEMS

3.6.1.2 Primary Containment Air Lock

LCO 3.6.1.2 The primary containment air lock shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

1. Entry and exit is permissible to perform repairs of the air lock components.
2. Enter applicable Conditions and Required Actions of LCO 3.6.1.1, "Primary Containment," when air lock leakage results in exceeding overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One primary containment air lock door inoperable.	------------NOTES------------ 1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the air lock are inoperable and Condition C is entered. 2. Entry and exit is permissible for 7 days under administrative controls. -----------------------------	
	A.1 Verify the OPERABLE door is closed. AND	2 hours
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Lock the OPERABLE door closed.	24 hours
	AND	
	A.3 --------NOTE--------- Air lock doors in high radiation areas or areas with limited access due to inerting may be verified locked closed by administrative means. --------------------	
	Verify the OPERABLE door is locked closed.	Once per 31 days
B. Primary containment air lock interlock mechanism inoperable.	------------NOTES------------ 1. Required Actions B.1, B.2, and B.3 are not applicable if both doors in the air lock are inoperable and Condition C is entered. 2. Entry into and exit from primary containment is permissible under the control of a dedicated individual. ----------------------------	
	B.1 Verify an OPERABLE door is closed.	2 hours
	AND	
		(continued)

PLANT OPERATING MANUAL

VOLUME I
BOOK 2

ADMINISTRATIVE INSTRUCTION

UNIT
0

0AI-107

INSTRUCTIONS FOR WORKING IN HOT ENVIRONMENTS

REVISION 10

EFFECTIVE DATE

02/15/99

Sponsor	Signature and Date on File Industrial Hygiene and Safety Representative	Date
Approval	Signature and Date on File Outages and Scheduling Manager	Date

REVISION SUMMARY

Removal of reference to LPU and change in signature authority.

LIST OF EFFECTIVE PAGES

Page(s)	Revision
1-20	10

TABLE OF CONTENTS

1.0 PURPOSE

The purpose of this procedure is to provide guidance to all employees for preventing heat-induced occupational illnesses or injuries, thus, enhancing employee safety and increasing productivity.

Heat related fatigue can lead to decreased job performance as well as contributing to work place accidents and illness. Productivity and worker safety can be enhanced through the management of heat stress.

This program is based EPRI Report NP-4453 "Heat Stress-Management Program for Nuclear Power Plants". The EPRI report outlines a three Step method for managing heat stress. These steps are: environmental assessment by trained evaluators, control methods, and training.

2.0 REFERENCES

2.1 EPRI NP 4453, Heat Stress Management Program for Nuclear Power Plants

2.2 NIOSH Publication 72-10269, Criteria for a Recommended Standard: Occupational Exposure to Hot Environments

2.3 NIOSH, The Industrial Environment - Its Evaluation and Control, Chapters 30, 31, and 38

2.4 The American Industrial Hygiene Association, Heating and Cooling for Man in Industry

2.5 E. Kamon and C. Ryan, Effective Heat Strain Index Using Pocket Computer, AIHA Journal, August 1981

2.6 American Red Cross, Advanced First Aid and Emergency Care, Second Edition

2.7 E&RC-0136, Setup and Use of Airline Respiratory Protection Devices

2.8 E&RC-0229, Control & Use of HEPA Vacuum Cleaners and Mobile Air Filtration Units

3.0 DEFINITIONS

3.1 Heat Stress

The physiological stress which occurs when the body's temperature rises above normal. This occurs when the body produces or gains more heat than it is capable of losing. It is caused by any combination of air temperature, thermal radiation, humidity, air flow, restrictive clothing, and physical work load which may result in elevated core body temperature and subsequent illness.

3.2 Action Time

An estimate of the length of time workers may be exposed in hot environments and not suffer heat stress disorders, used for planning purposes. The length of Action Times is not absolute because of worker variability in response to heat. The times reflect an approximate 2°F rise in body temperature.

3.3 Protective Clothing (PCs)

Items worn to prevent radioactive contamination.

3.4 Wet Suit

Full body impermeable plastic suit worn to prevent radioactive skin contamination.

3.5 Chemical Suit

Full body impermeable neoprene or Tyvek coveralls worn to prevent chemical skin contamination.

3.6 Personal Cooling Device

Equipment such as ice vests or vortex cooling units placed on a person to minimize heat gain and/or increase heat loss.

3.7 Supplied Air Hood/Helmet

Air-supplied hood respirator which delivers respirator air over the head and body.

3.0 DEFINITIONS

3.8 WBGT

Wet Bulb Globe Thermometer - used to establish the work area Temperature Index Heat Stress that allows for the effects of Humidity, and Radiant Heat, that modify dry bulb temperatures.

3.9 Acclimation

The gradual process of improved heat tolerance after continuous exposure to heat. Acclimation consists of reduced heart rate, increased sweat production, production of less salty sweat, and lower body temperature.

3.10 Dry Bulb Temperature

The temperature as measured by a standard thermometer without respect to humidity or radiant heat.

3.11 Globe Temperature

Temperature resulting from radiant heat sources, measured with a black globe thermometer.

3.12 High Heat Stress Job/Work

Any job in which the calculated Action Time is less than 30 minutes.

3.13 Metabolic Heat Load

Heat generated from physical work (muscle contraction).

3.14 Moderate Heat Stress Job/Work

Any job/work in which the calculated Action Time is greater than 30 minutes but less than 240 minutes.

3.15 Relative Humidity

The amount of moisture in the air compared to the amount of moisture the air can hold for a given temperature.

3.0 DEFINITIONS

3.16 Recovery Period

Recovery time allocated to workers who have performed work in hot environments. Recovery shall not take place in a hot environment. Water should be available for consumption in the recovery area.

3.17 Self-Determination

Allowing for worker discretion to exit High Heat Stress Work Areas when he/she feels the onset of heat stress symptoms.

3.18 Time Keeper

A person responsible to monitor action times.

3.19 Wet-Bulb Temperature (natural)

The temperature of the air when it is subjected to evaporative cooling.

3.20 High Temperature Work Level

Any work area > 95°F.

3.21 Designated Drinking Areas (DDA)

Specific areas designated within the Radiation Control Area to allow ingestion of liquids as part of the Heat Stress Program.

4.0 RESPONSIBILITIES

4.1 General Manager - Brunswick Plant

The General Managers - Brunswick Plant are responsible for the implementation of this procedure to ensure that personnel who perform work in high temperature environments follow the guidance of this procedure.

4.0 RESPONSIBILITIES

4.2 Managers

4.2.1 Managers will ensure that the supervisors reporting to them utilize this procedure and follow its guidance when planning work in hot environments.

4.2.2 Managers shall ensure that training or instruction on heat stress mitigation is arranged for and conducted for employees prior to initial work in high temperature environments.

4.3 Supervisor

4.3.1 The supervisor or person in charge of the job is responsible for following the guidance in this procedure when planning a job that is to be performed within a hot environment, and ensure that heat stress mitigation has been considered during job planning.

4.3.2 Safety of employees shall be the responsibility of supervision whenever employees must enter or work in a hot environment or may be subject to heat stress causing conditions.

4.3.3 Shall ensure that heat stress caused illnesses are recorded on the SAF-CPL-009 form as appropriate.

4.4 Individuals

Each individual is responsible for complying with:

4.4.1 The requirements of this procedure.

4.4.2 Written and/or oral instructions given by supervision on mitigating heat stress.

4.4.3 Instructions given by the supervisor on the use of body cooling devices.

4.4.4 Being attentive to symptoms of heat stress while working in hot environments, and stopping work and notifying their supervisor if they feel ill due to heat stress.

4.4.5 Each individual is responsible for being prepared to work in a hot environment; rested and have no medical problems that would be affected by heat related work.

4.0 RESPONSIBILITIES

4.5 Heat Stress Evaluators

Individuals trained in the use of this procedure and the WBGT thermometer for the purpose of evaluating the potential for heat stress during jobs completed on site.

4.6 Industrial Hygiene/Safety Representative

Shall provide technical assistance on plant heat stress issues.

4.7 Training Department

Is responsible for teaching heat stress in initial GET and in the annual retraining.

5.0 INSTRUCTIONS

5.1 Precautions

CAUTION

Workers should never work alone in high heat stress areas.

5.1.1 If any individual begins to feel symptoms of heat illness, he/she shall immediately exit the area, de-suit, notify the job supervisor, rest in a cool area, and drink plenty of fluids. Seek medical help if necessary, by calling the Control Room (extension 4444).

5.1 Precautions

5.1.2 All jobs in high temperature environments should address heat stress prevention controls in the planning stages.

1. In situations where individuals know that their work schedule for the next day will involve entering a heat stress area, they should drink plenty of liquids in the 24 hours prior to reporting to work.

2. Action Times, recovery times, personnel rotation, and the use of body cooling devices should be addressed.

3. Whenever possible, engineering controls should be used to eliminate/reduce the exposure (i.e., isolation of the heat source, introduction of cooled air, circulation of present air, reduced humidity, etc.). The impact of these engineering controls should be reviewed with the Environmental and Radiation Control (E&RC) unit for jobs in radiologically controlled areas. When introducing air circulation and handling devices, follow the guidelines provided in E&RC-0229.

5.1.3 Individuals who work in hot environments may become dehydrated due to sweating. Water should be replaced at rest breaks to prevent heat-related illness. Employees should be encouraged to drink small amounts often, regardless of thirst. Salt tablets are <u>not</u> recommended. Liquids designed to replace these salts (i.e., Gatorade or water) are recommended as replacement fluids.

5.1.4 Individuals who work in high temperature environments must periodically rest in a cooler area to shed body heat. Duration of breaks, extent of clothing removal, and rest area should be determined by the job supervisor, using the guidance in Section 5.2. Certain employees may require varying rest periods with some requiring less time than shown in Section 5.2.

5.1.5 Individuals who will be working in high temperature environments for the first time will be more susceptible to heat illness than those accustomed to hot work. After working in hot environments for several days, their bodies may adjust to heat exposure and they may tolerate longer heat exposures at higher work rates (acclimatization).

5.1 Precautions

5.1.6 Individuals vary greatly in their tolerance to heat exposure. Factors which may affect heat tolerances may include:

- Age
- Weight
- Sex
- Physical fitness
- General health
- Colds, viruses, and infection
- Some medications
- Consumption of alcoholic beverages

5.2 Heat Illness Prevention and First Aid

> **NOTE:** The following first aid actions are recommendations only. If any individual begins to feel symptoms of heat illness, the worker should immediately exit the area, notify the supervisor, and seek first aid/medical attention.

5.2.1 Workers should be encouraged to drink one pint of water/fluid per hour of scheduled work prior to entering high heat areas.

5.2.2 Workers shall be encouraged to drink water/fluid after high temperature work to maintain fluid balance.

5.2.3 Where feasible, high temperature work shall be scheduled to minimize thermal stress in the work area. This includes scheduling work at times where the WGBT and or metabolic heat load are lower and or anti-C requirements are less restrictive.

5.2 Heat Illness Prevention and First Aid

5.2.4 Prior to re-entering a High Temperature Work area, workers shall have an adequate recovery period to dissipate excess heat and replace water. Recovery shall take place in a cool location (less than 80 degrees F) where drinking water is available.

The length of the recovery period depends on the length of exposure and the maximum stay time of the job. Recovery periods of up to one hour may be necessary for jobs which approach or exceed the planned stay times. The following formula shall be used as a general guide for determining the minimum length of recovery period. Actual recovery time can be modified only by agreement of both worker and supervisor. This shall be documented on Attachment 4, Heat Stress Evaluation Form.

> **NOTE:** All times used should be in minutes
>
> $$REC = \frac{AET \times 60}{MST}$$
>
> REC ---------- Recovery Time
> AET ---------- Actual Exposure Time to the Hot Environment
> MST ---------- Actual Stay Time or Action Time from Attachment 2

5.2.5 Heat Cramps - are muscle spasms due to a loss of salt through sweating. The legs, arms, and abdominal muscles are the most commonly affected muscle groups. Cramps can also result from drinking large amounts of water without electrolytes. Heat cramps may be a sign of approaching heat exhaustion.

First Aid - Rest in cool area, drink water or liquids containing electrolytes and eat food high in salt content.

5.2.6 Heat Exhaustion - is dehydration caused by prolonged heavy sweating. There is insufficient flow of blood to the brain (blood is shunted to the skin to lose heat). Symptoms include dry mouth, excessive thirst, loss of coordination, headache, dizziness, fatigue, pale and shaky look, and cool clammy skin. This condition may develop into heat stroke.

First Aid - Rest in a cool area, lie down, elevate feet, apply cool wet clothes, fan with air, and drink liquids.

5.2 Heat Illness Prevention and First Aid

5.2.7 Heat Stroke - is a serious medical emergency caused by a failure of the body's cooling mechanisms. Symptoms include hot, dry skin (sweating stops), extremely high body temperatures, chills, convulsions, and unconsciousness.

First Aid: Immediate, rapid cooling of the body is necessary. Use safety showers, move air over the body with a fan or by fanning, or cover the body with a wet sheet. Call the Control Room (extension 4444) to seek immediate medical attention.

5.3 Use of Recommended Action Times

5.3.1 A fundamental rule of heat stress management is that self-determination by the worker should take precedence over other factors. Attachment 2, Recommended Action Times, may be modified and even exceeded only by agreement of supervisor and worker. Attachment 4, Heat Stress Evaluation Form, shall document this change, however a worker must leave a high temperature work area if he/she feels the onset of heat stress symptoms.

5.3.2 By using the recommended Action Time as a general guideline, and assessing the physical condition of his workers, the job supervisor can determine how long his workers may be able to work before rest breaks are given. Workers must, and have the right to, exit the hot environment prior to the time limit if they feel that they cannot continue.

5.3.3 Work should be planned so that an adequate number of workers are prepared to work in the high temperature environment. The supervisor should also consider whether there are enough workers to complete the task if some workers cannot last the recommended time.

5.3.4 A worker may extend his Action Time long enough to bring the task to a satisfactory and safe stopping point if he/she feels fully capable of staying longer and has supervision approval. In no case should this extended time be more than 25% above the recommended time limit stated in Attachment 4.

5.4 Heat Stress Evaluation

5.4.1 The heat stress evaluation process involves assessing the variables that affect heat stress, including WBGT measurements, metabolic work load, and clothing type. These factors are converted to recommended action times for planning purposes. A Heat Stress Evaluation Form (Attachment 4), or other record containing the same information should be used for heat stress job planning.

5.4.2 All potential High Heat Stress Work shall be identified.

5.4.3 For initial evaluations, the WBGT shall be measured using a WBGT Meter. Measurements shall be representative of the work area thermal load. Succeeding evaluations may be based solely on dry bulb temperature when a correlation between dry bulb temperature and WBGT is established.

> **NOTE:** Care must be used when conducting WBGT readings so as to not create an ALARA concern for the rare case of a worker being assigned both a dose and a heat stress action time; the most limiting shall be used.

5.4.4 The type of work clothing required for the job shall be determined. The categories include: street clothing, single cotton blend or paper/Tyvek coveralls, double cotton blend coveralls, and single cotton blend coveralls with impervious plastic (rain suit) or Tyvek outer suit.

5.4.5 The metabolic heat load shall also be assessed using Attachment 1 as a guide.

5.4.6 The Job Action Time is determined from Attachment 2 using the WBGT reading, clothing type, and metabolic load. Action Times are used for job planning. Action Times stated in Attachment 2 are <u>not</u> absolute because of the great variability in worker response to heat stress. Many healthy/acclimatized workers could exceed the Action Times without suffering any adverse effects. However, a few workers could experience heat stress symptoms prior to reaching the maximum Action Time.

5.4.7 A re-evaluation is necessary, if there are changes in the WBGT (+/-3°F), the metabolic work load category or the required clothing type during the course of the operation.

5.5 Use of Ice Vests

5.5.1 By using Attachment 2, the job supervisor can determine if ice vests would be beneficial for the job.

1. Proper handling of ice vests and ice packs is essential to maintaining an adequate supply in good condition. The flow path (Attachment 3) for the use of the vests must be followed by all workers to ensure availability.
2. Job supervisors shall ensure their workers are trained in the use of ice vests prior to using the vests in hot environments.
3. The ice vest should be worn so that the vest fits snugly. A shirt should be worn under the vest to prevent frost burn. Under garment shirts are available upon request.
4. As much as practical, the ice vest should not be donned until just prior to entering the hot environment.
5. The ice vest will provide cooling only while the ice is melting. Once the ice has melted, body temperature will increase quickly. Workers should monitor their condition and exit the work area as soon as the ice vest has lost its cooling effectiveness.

5.6 Use of Supplied Air Hood/Helmets

Supplied air hood/helmets are used mainly as respirators and their uses are authorized only by the E&RC Unit.

5.7 Designated Drinking Areas

> **NOTE:** DDA's are only a part of the total Heat Stress Program. Control of the Areas to ensure that sanitary conditions exist and radiological controls are followed will dictate that the number and locations are limited.

5.7.1 RC Supervision will evaluate the request for DDA's authorize their placement and determine survey requirements.

5.7 Designated Drinking Areas

5.7.2 Areas will be bounded off and posted similar to the following: Designated Drinking Area

1. Whole Body Frisk Required Prior to Entry
2. Workers Shall Drink Only Within the DDA
3. "NO ANTICONTAMINATION CLOTHING ALLOWED"

5.7.3 Personnel **SHALL NOT ENTER** the **DDA** dressed in, or with anticontamination clothing.

5.7.4 Personnel **SHALL PERFORM** a **WHOLE BODY FRISK PRIOR TO ENTRY** and **DRINKING** in a **DDA**.

Work Rate Guidelines

CATEGORY	TYPE OF ACTIVITY	EXAMPLES
LIGHT	- sitting with moderate arm and trunk movement - sitting with moderate arm and leg - standing, light work at machine or bench - standing, light work with some walking and minimal climbing	- inspections and surveys with minimal climbing - supervising or monitoring areas or equipment - bench work
MODERATE	- standing with moderate work - walking with moderate lifting or pushing - walking with occasional ladder or stair climbing	- painting - floor cleaning - insulation removal or installation - fitting and welding light pieces - surveys and inspections with moderate climbing
HEAVY	- walking with frequent stair and ladder climbing - heavy lifting, pushing, or pulling	- scaffold erection - rigging - transporting equipment by hand - manual decontaminating - shoveling - mopping

ATTACHMENT 2
Page 1 of 1
Recommended Action Times*

	Single Cotton Blend or Paper Coveralls			Double Cotton Blend or Paper Coveralls			Single Cotton Blend Plus Impervious Garment			
WBGT (F°)	LIGHT WORK	MODERATE WORK	HEAVY WORK	LIGHT WORK	MODERATE WORK	HEAVY WORK	LIGHT WORK	MODERATE WORK	HEAVY WORK	SUPPLIED AIR HOOD/ HELMETS (HOURS)
75-78.9	NL	NL	150m	NL	180m	90m	190m	65m	40m	4
79-82.9	NL	145m	80m	240m	80m	50m	130m	45m	30m	4
83-86.9	225m	75m	45m	165m	55m	35m	90m	35m	20m	4
87-90.9	150m	50m	35m	105m	40m	25m	55m	30m	15m	4
91-93.9	105m	40m	25m	80m	35m	20m	45m	25m	15m	3
94-97.9	75m	35m	15m	50m	25m	15m	35m	20m	PCR	3
98-100.9	50m	25m	PCR	45m	20m	PCR	25m	15m	PCR	3
101-104.9	35m	20m	PCR	30m	15m	PCR	20m	PCR	PCR	2 1/2
105-108.9	25m	15m	PCR	25m	PCR	PCR	15m	PCR	PCR	2 1/2
109-111.9	20m	PCR	PCR	20m	PCR	PCR	PCR	PCR	PCR	2 1/2
112-115.9	15m	PCR	PCR	15m	PCR	PCR	PCR	PCR	PCR	2

NOTE: Ice vests will provide cooling for 45 to 120 minutes depending on conditions. Work time limits should be determined by the user based on when his/her ice vest no longer provides cooling.

m = minutes h = hours NL = No Limit PCR = Personal Cooling Recommended

*Without Personal Cooling Equipment

ATTACHMENT 3
Page 1 of 1

Cool Vest Flow Path

*May reuse vest with fresh ice packs

Vest in Bin
Acquired By User

Used in
Cont. Area
↓
Leaves Area
↓
Whole Body Frisk
with Vest on

Vest Clean*
↓
Remove and
frisk ice pack

Ice Pack
Clean
↓
Return ice
pack
to freezer

Ice Pack
Cont.
↓
Put ice
pack in
Yellow Bag

Monitors vest
with SAM or
other frisking devices
↓
Place vest in Vest
Laundry Barrel
or other
designated location
↓
Vests laundered
and frisked
↓
Vests put in bins

Vest Cont.
↓
Put vest and
ice in a
Yellow Bag
↓
Return Cont.
ice packs and
vests to
Personnel
Decon
or other
designated
location

Used in
Clean Area
↓
Leaves Area*
↓
Returns Ice Pack
to Freezer
↓
Monitors Vest
With SAM or other
frisking devices
↓
Places vest in Vest
Laundry Barrel or
Other Designated
Location

Heat Stress Evaluation Form
(Section 5.4)

Task(s): ____

____

Supervisor: ______________________ Job Date: __________

Job Location: ____

Number of Workers: ______________ Est. Person-Hours: __________

Plant Status (for job planning use): ______________________

5.2.4 Modified Action Time/Recover Period (**AS REQUIED**)
Signatures (Worker)______________________
(Supervisor)______________________
(Time Keeper)______________________

5.4.4 CLOTHING TYPE
(Circle)

single coveralls	double coveralls	impervious outer &
(cotton blend) or	(cotton blend) or	cotton blend inner
paper coveralls	paper coveralls	

5.4.5 METABOLIC HEAT LOAD
(Circle)

low	moderate	high

5.4.3 Dry Bulb = __°F Wet Bulb = ____°F Globe Temp = __°F WBGT = ____°F

5.4.6 ACTION TIME = _______ minutes 5.2.4 Recovery Period = _______minutes

CONTROL METHODS:____

____

____

____

Signature (Evaluator): ______________________ Date: __________

Signature (Job Supervisor): ______________________ Date: __________

EOP-01-UG
Attachment 6
Reactor Water Level Caution
(Caution 1)

ATTACHMENT 6
REACTOR WATER LEVEL CAUTION
(Caution 1)

A reactor water level instrument may be used to determine reactor water level only when the conditions for use as listed in Table 1 are satisfied for that instrument.

TABLE 1

CONDITIONS FOR USE OF REACTOR WATER LEVEL INSTRUMENTS

NOTE

Reference leg area drywell temperature is determined using Figure 13, ERFIS, or Instructional Aid based on Figure 13.

NOTE

If the temperature near any instrument run is in the UNSAFE region of the REACTOR SATURATION LIMIT (Figure 14), the instrument may be unreliable due to boiling in the run.

NOTE

Immediate reference leg boiling is not expected to occur for short duration excursions into the unsafe region due to heating of the drywell. The thermal time constant associated with the mass of metal and water in the reference leg will prohibit immediate boiling of the reference leg. Reference leg boiling is an obvious phenomenon. Large scale oscillations of all water level instruments associated with the reference leg that is boiling will occur. This occurrence will be obvious and readily observable by the operator. Additionally, if the operator is not certain whether boiling has occurred, he can refer to plant history as provided on water level recorders or ERFIS. Reference leg boiling is indicated by level oscillations without corresponding pressure oscillations.

Instrument	Conditions for Use
Narrow Range Level Instruments C32-LI-R606A, B, C (N004A, B, C) C32-LPR-R608 (N004A, B) Indicating Range 150-210 Inches Cold Reference Leg	<u>Unit 1 Only</u>: The indicated level is in the SAFE region of Figure 15. <u>Unit 2 Only</u>: The indicated level is in the SAFE region of Figure 15A.
Shutdown Range Level Instruments B21-LI-R605A, B (N027A, B) Indicating Range 150-550 Inches Cold Reference Leg	The indicated level is in the SAFE region of Figure 16. NOTE To determine reactor water level at the Main Steam Line Flood Level (MSL), see Figure 21. NOTE Figure 21 has two curves: The upper curve is for reference leg area drywell temperature equal to or greater than 200°F. The lower curve is for reference leg area drywell temperature less than 200°F.

ATTACHMENT 6 (Cont'd)

TABLE 1 (Cont'd)

Instrument	Conditions for Use
Wide Range Level Instruments B21-LI-R604A, B (N026A, B) C32-PR-R609 (N026B) Indicating Range 0-210 Inches Cold Reference Leg	1. Temperature on the Reactor Building 50' below 140°F (B21-XY-5948A A2-4, B21-XY-5948B A2-4, ERFIS Computer Point B21TA102, OR B21TA103) AND 2. IF the reference leg area drywell temperature is in the UNSAFE region of the Reactor Saturation Limit (Figure 14), THEN the indicated level is greater than 20 inches OR IF the reference leg area drywell temperature is in the SAFE region of the Reactor Saturation Limit (Figure 14), THEN the indicated level is greater than 10 inches.

ATTACHMENT 6 (Cont'd)

TABLE 1 (Cont'd)

Instrument	Conditions for Use
Fuel Zone Level Instruments B21-LI-R610 (N036) B21-LR-R615 (N037) Indicating Range -150 - +150 Inches Cold Reference Leg	1. IF the reference leg area drywell temperature is less than 440°F, THEN the indicated level is greater than -150 inches OR IF the reference leg area drywell temperature is greater than or equal to 440°F, THEN the indicated level is greater than -130 inches. AND 2. Reactor Recirculation Pumps are shutdown. NOTE To determine reactor water level at TAF, see Unit 1 Only: Figure 17 and Unit 2 Only: Figure 17A To determine reactor water level at the minimum steam cooling level (LL-4), see Unit 1 Only: Figure 18 and Unit 2 Only: Figure 18A To determine reactor water level at the minimum zero injection level (LL-5), see Unit 1 Only: Figure 19 and Unit 2 Only: Figure 19A To determine reactor water level at 90 inches, see Figure 20. Continued on next page.

ATTACHMENT 6 (Cont'd)

TABLE 1 (Cont'd)

Instrument	Conditions for Use
	<u>NOTE</u> Each figure has two curves: The upper curve for reference leg area drywell temperature greater than 200°F. The lower curve for reference leg area drywell temperature less than or equal to 200°F. If containment conditions are such that reference leg area temperatures could not be controlled and maintained less than the 200°F requirement, then the upper lines on the graph should be utilized. <u>NOTE</u> These level instruments are valid for indication with RHR LPCI flow.

ATTACHMENT 6 (Cont'd)

FIGURE 13
LEVEL INSTRUMENT REFERENCE LEG AREA
DRYWELL TEMPERATURE CALCULATIONS

1. For all Level Instruments EXCEPT B21-LI-R605 A, B, (N027 A, B); the reference leg area drywell temperature is the highest of the following points:

Recorder

CAC-TR-4426-1B Point 1258-1 ______
CAC-TR-4426-1B Point 1258-3 ______
CAC-TR-4426-2B Point 1258-2 ______
CAC-TR-4426-2B Point 1258-4 ______

OR

Microprocessor

CAC-TY-4426-1 Point 5801 ______
CAC-TY-4426-1 Point 5803 ______
CAC-TY-4426-2 Point 5802 ______
CAC-TY-4426-2 Point 5804 ______

ATTACHMENT 6 (Cont'd)

FIGURE 13 (Cont'd)
LEVEL INSTRUMENT REFERENCE LEG AREA
DRYWELL TEMPERATURE CALCULATIONS

2. For Level Instruments B21-LI-R605A, B (N027A, B), the reference leg area drywell temperature is the highest of the following points:

Recorder

CAC-TR-4426-1A Point 1258-22 ______
CAC-TR-4426-1B Point 1258-3 ______
CAC-TR-4426-2A Point 1258-23 ______
CAC-TR-4426-2A Point 1258-24 ______
CAC-TR-4426-2B Point 1258-2 ______
CAC-TR-4426-2B Point 1258-4 ______

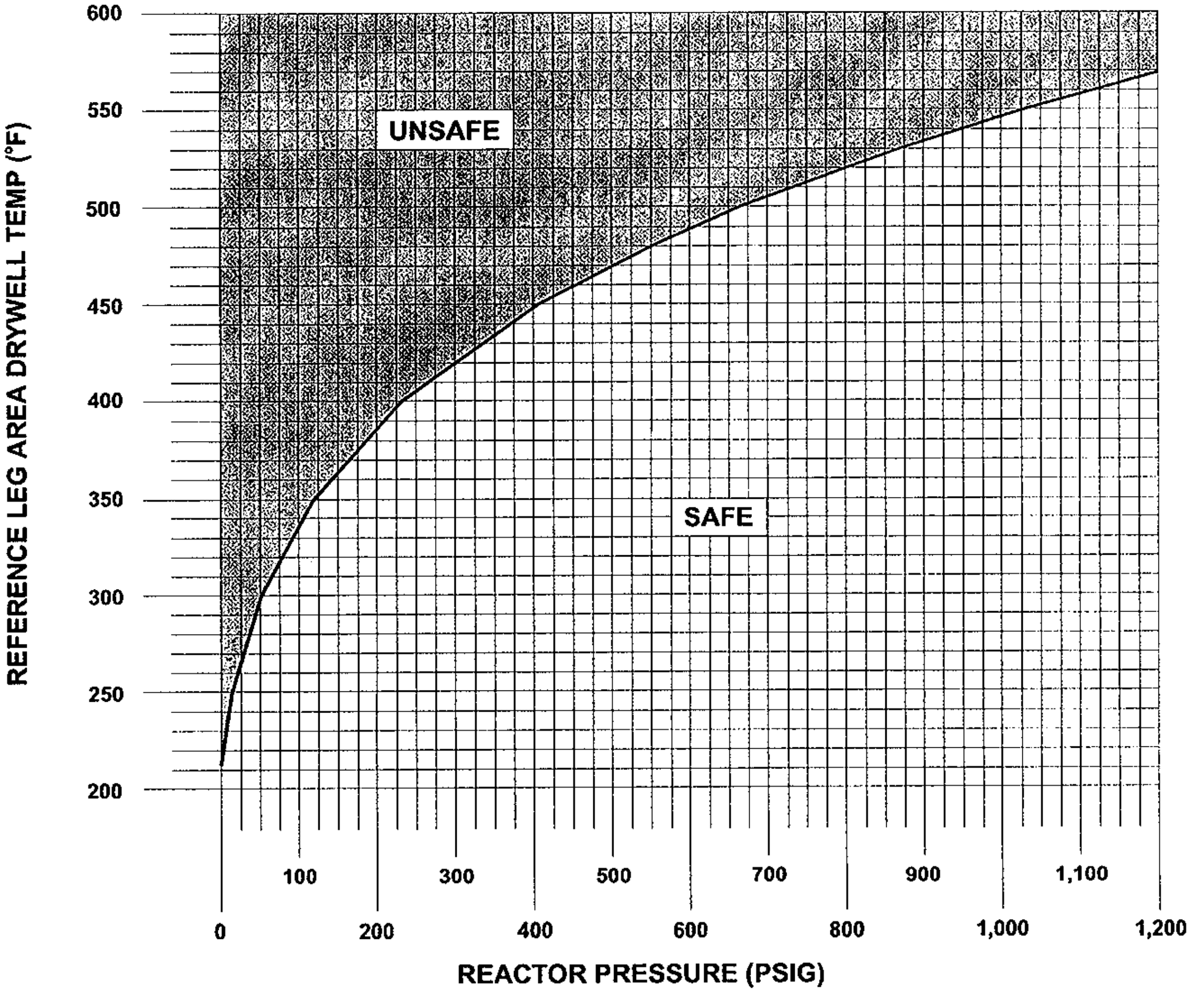
<u>OR</u>

Microprocessor

CAC-TY-4426-1 Point 5822 ______
CAC-TY-4426-1 Point 5803 ______
CAC-TY-4426-2 Point 5823 ______
CAC-TY-4426-2 Point 5824 ______
CAC-TY-4426-2 Point 5802 ______
CAC-TY-4426-2 Point 5804 ______

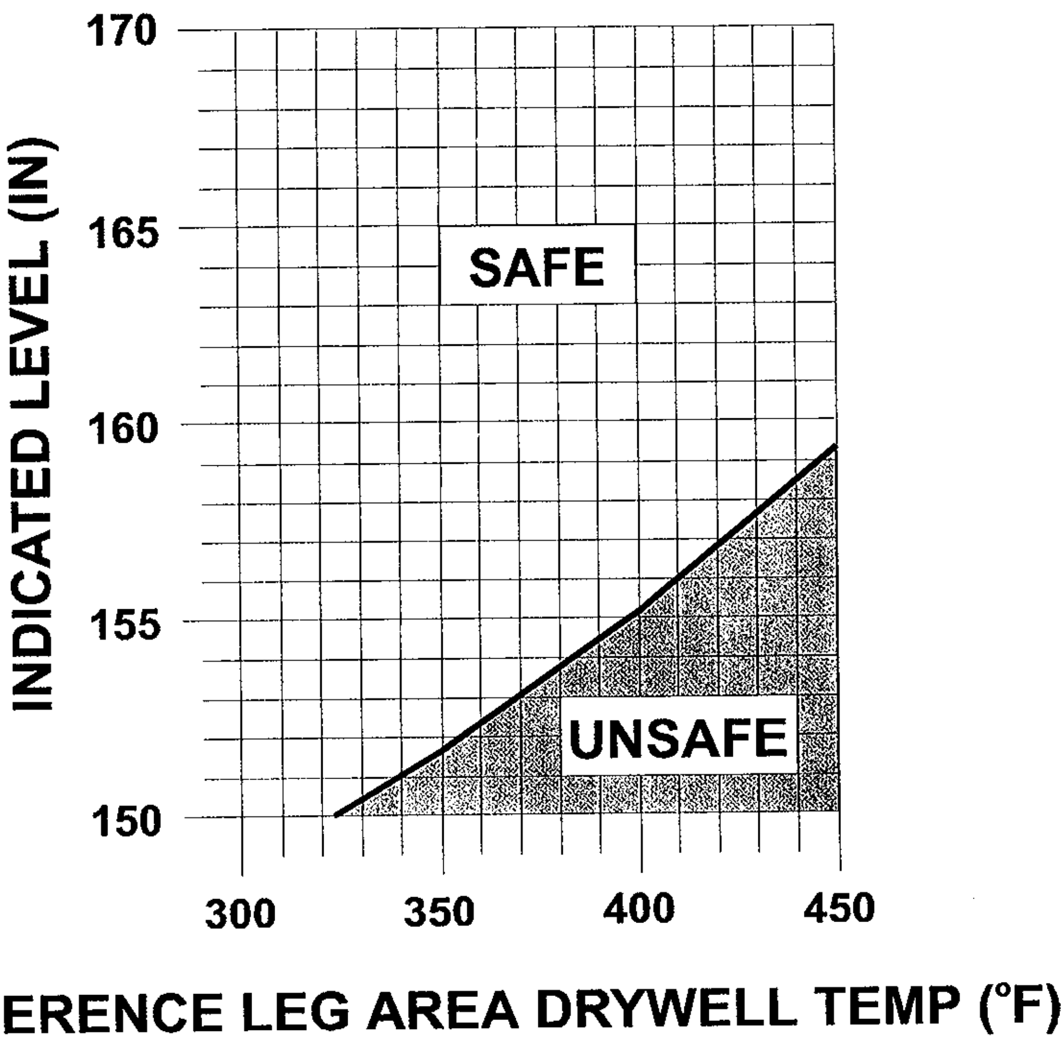
ATTACHMENT 6 (Cont'd)

FIGURE 14
REACTOR SATURATION LIMIT



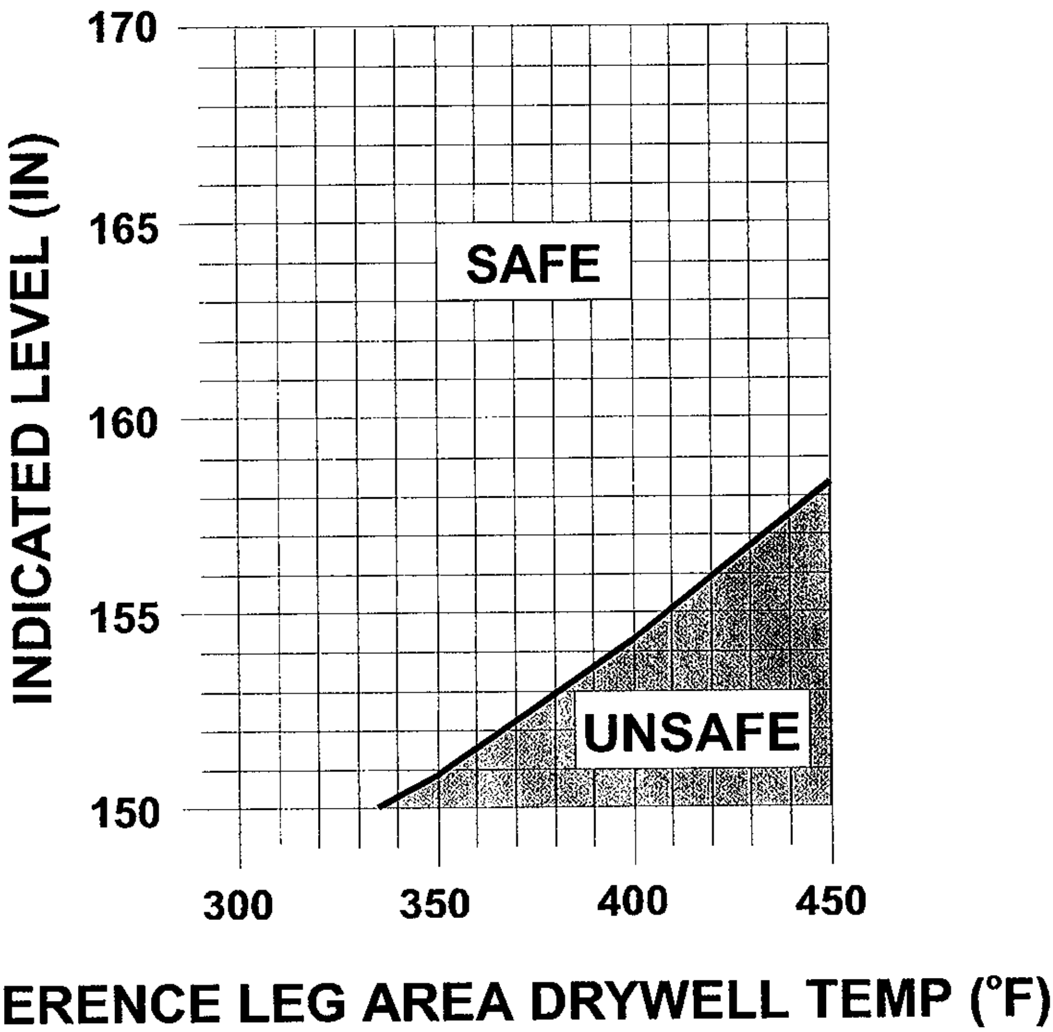
ATTACHMENT 6 (Cont'd)

FIGURE 15
UNIT 1 NARROW RANGE LEVEL
INSTRUMENT (N004A, B, C) CAUTION



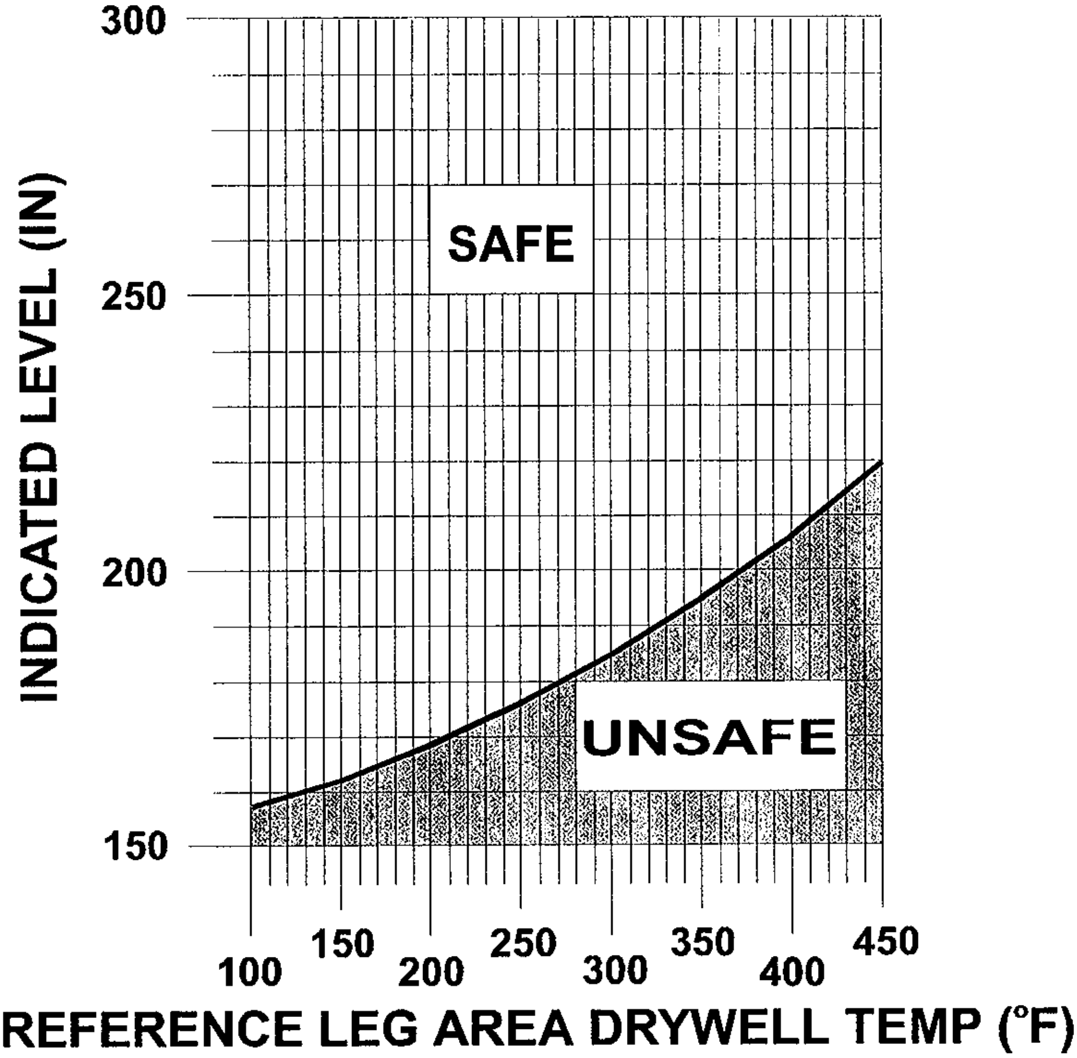
ATTACHMENT 6 (Cont'd)

FIGURE 15A
UNIT 2 NARROW RANGE LEVEL
INSTRUMENT (N004A, B, C) CAUTION



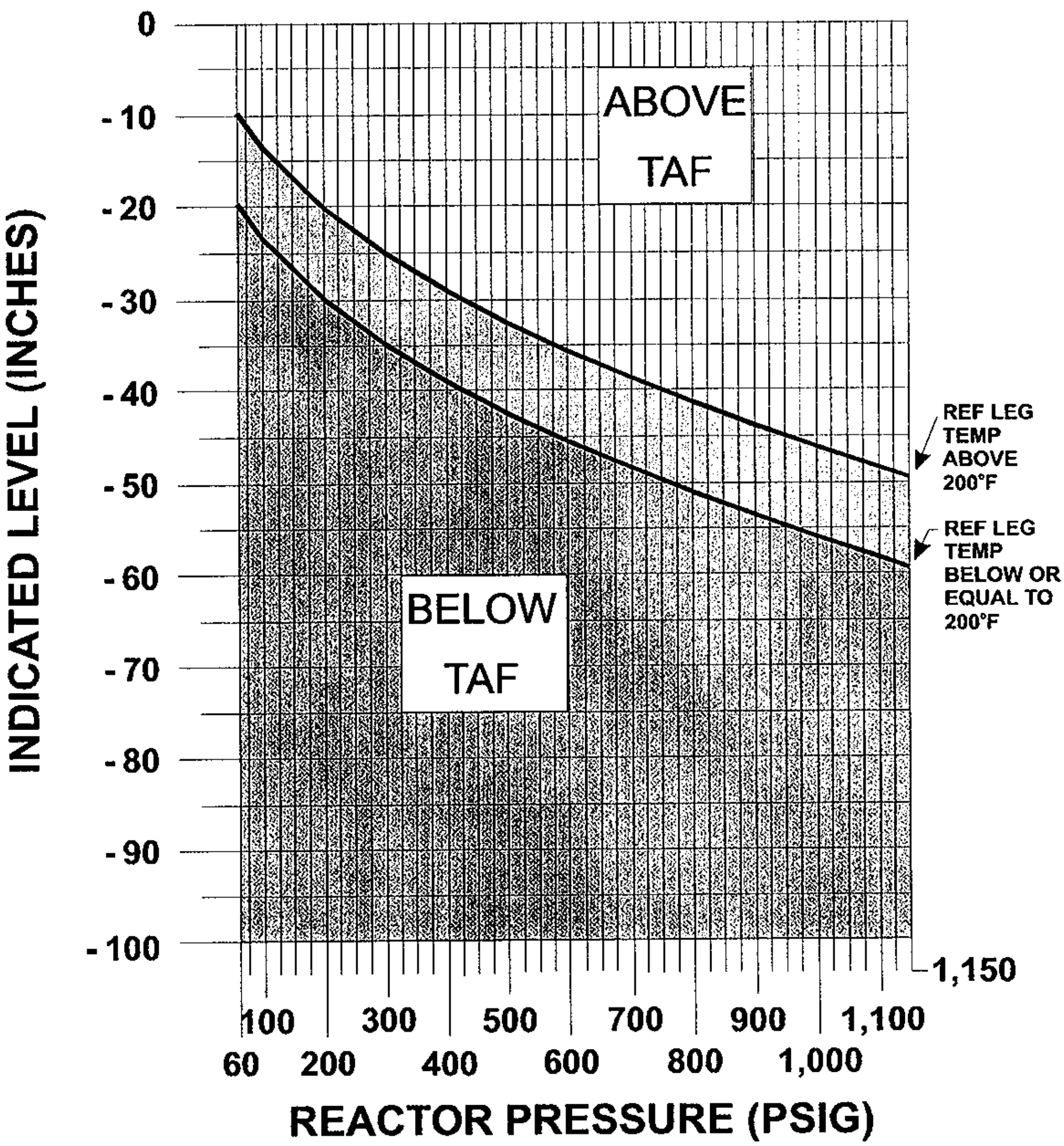
ATTACHMENT 6 (Cont'd)

FIGURE 16
SHUTDOWN RANGE LEVEL
INSTRUMENT (N027A, B) CAUTION



ATTACHMENT 6 (Cont'd)

FIGURE 17
UNIT 1 REACTOR WATER LEVEL AT TAF

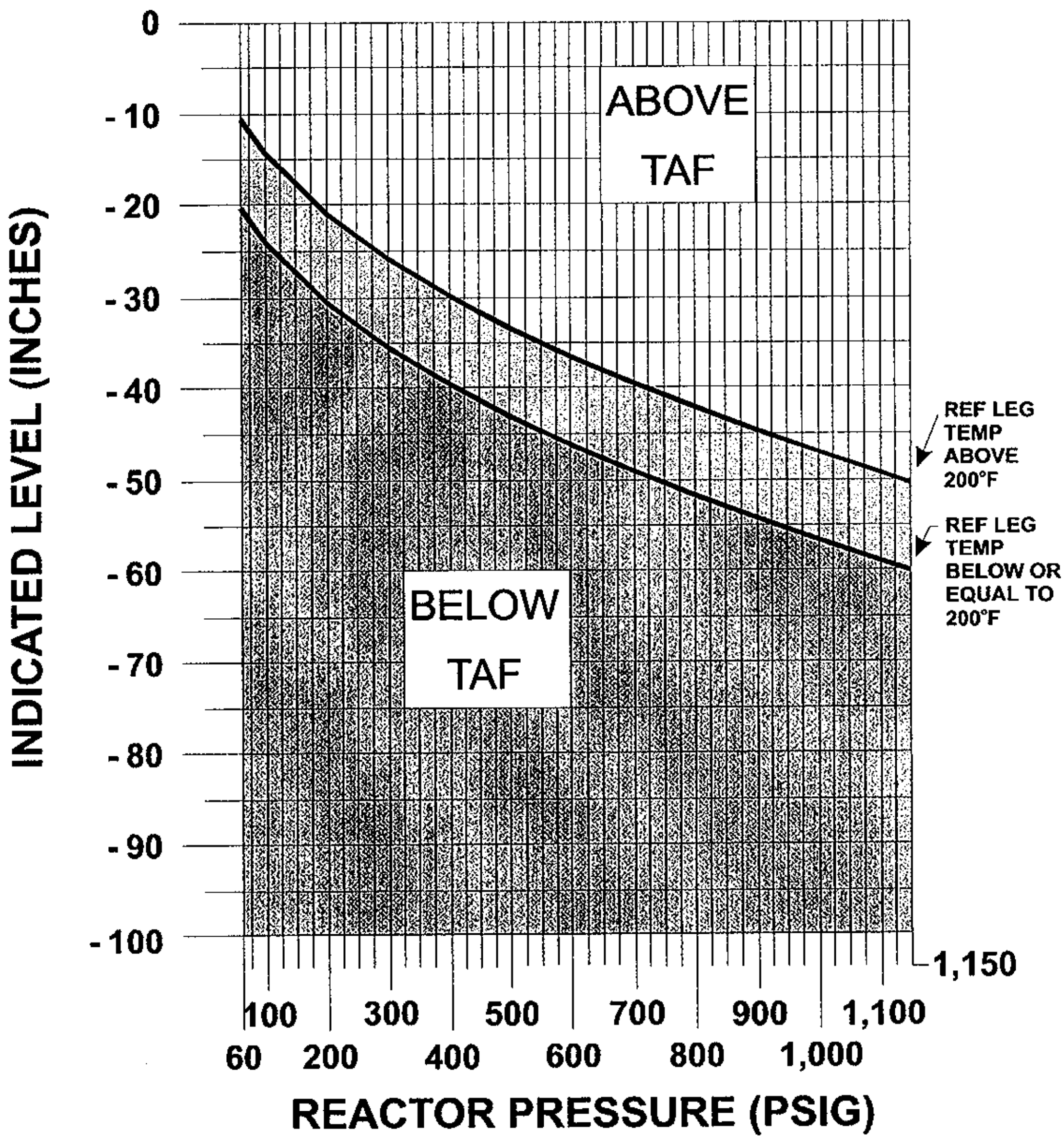


NOTE

WHEN REACTOR PRESSURE IS LESS THAN 60 PSIG, USE INDICATED LEVEL. TAF IS -7.5 INCHES.

ATTACHMENT 6 (Cont'd)

FIGURE 17A
UNIT 2 REACTOR WATER LEVEL AT TAF

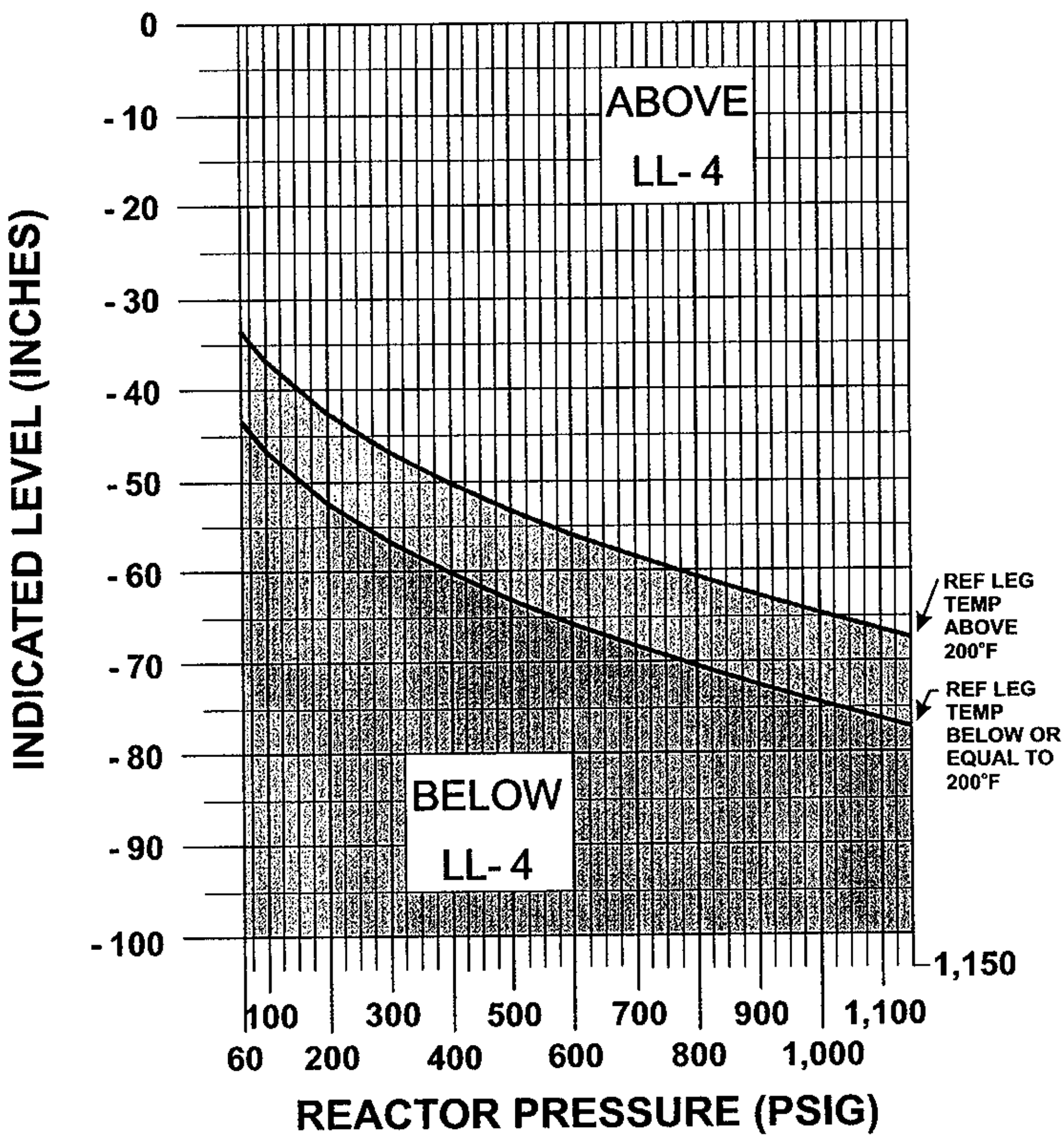


NOTE

WHEN REACTOR PRESSURE IS LESS THAN
60 PSIG, USE INDICATED LEVEL.
TAF IS -7.5 INCHES.

ATTACHMENT 6 (Cont'd)

FIGURE 18
UNIT 1 REACTOR WATER LEVEL AT LL-4
(MINIMUM STEAM COOLING LEVEL)

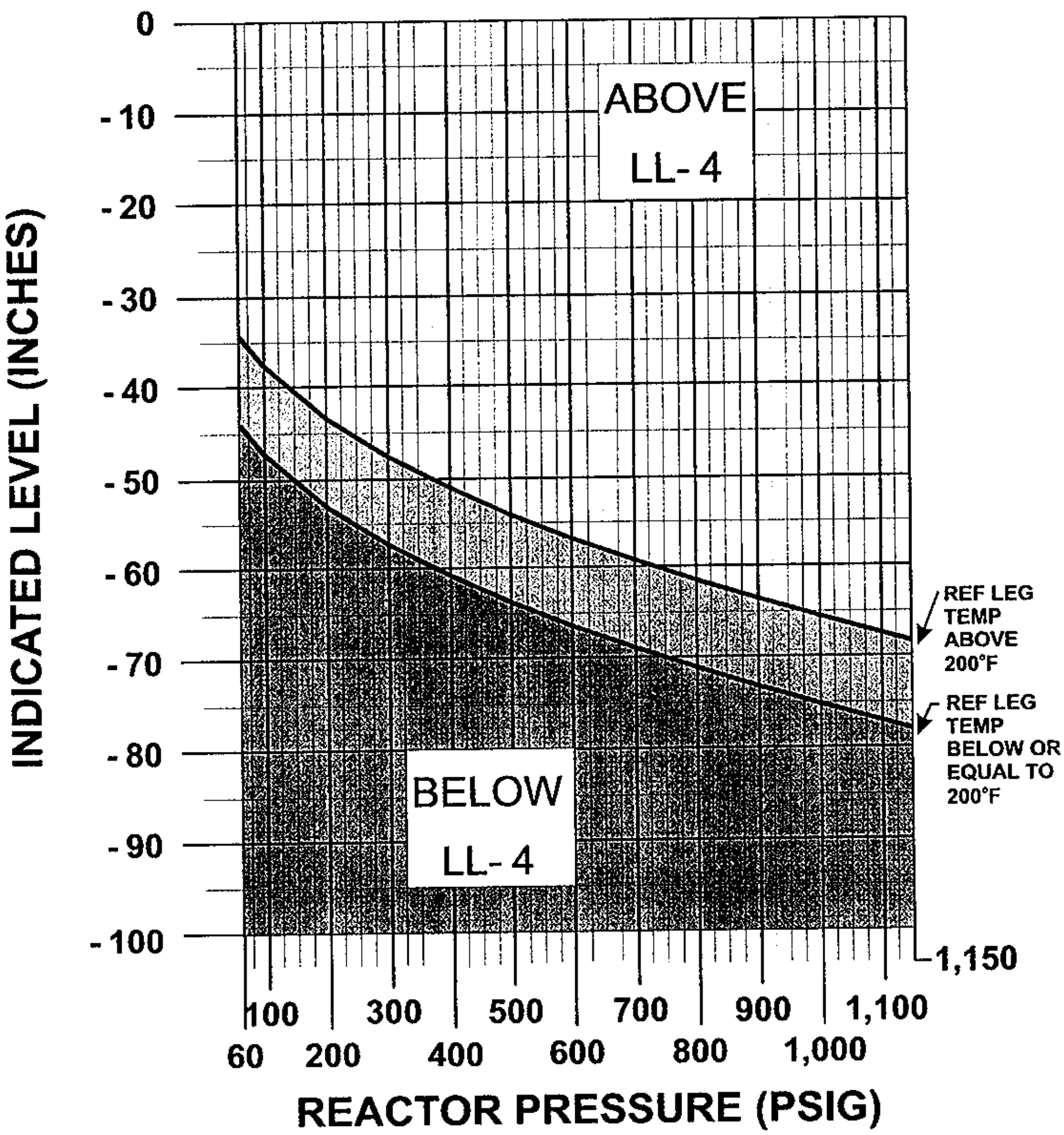


NOTE

WHEN REACTOR PRESSURE IS LESS THAN
60 PSIG, USE INDICATED LEVEL.
LL-4 IS -32.5 INCHES.

ATTACHMENT 6 (Cont'd)

FIGURE 18A
UNIT 2 REACTOR WATER LEVEL AT LL-4
(MINIMUM STEAM COOLING LEVEL)

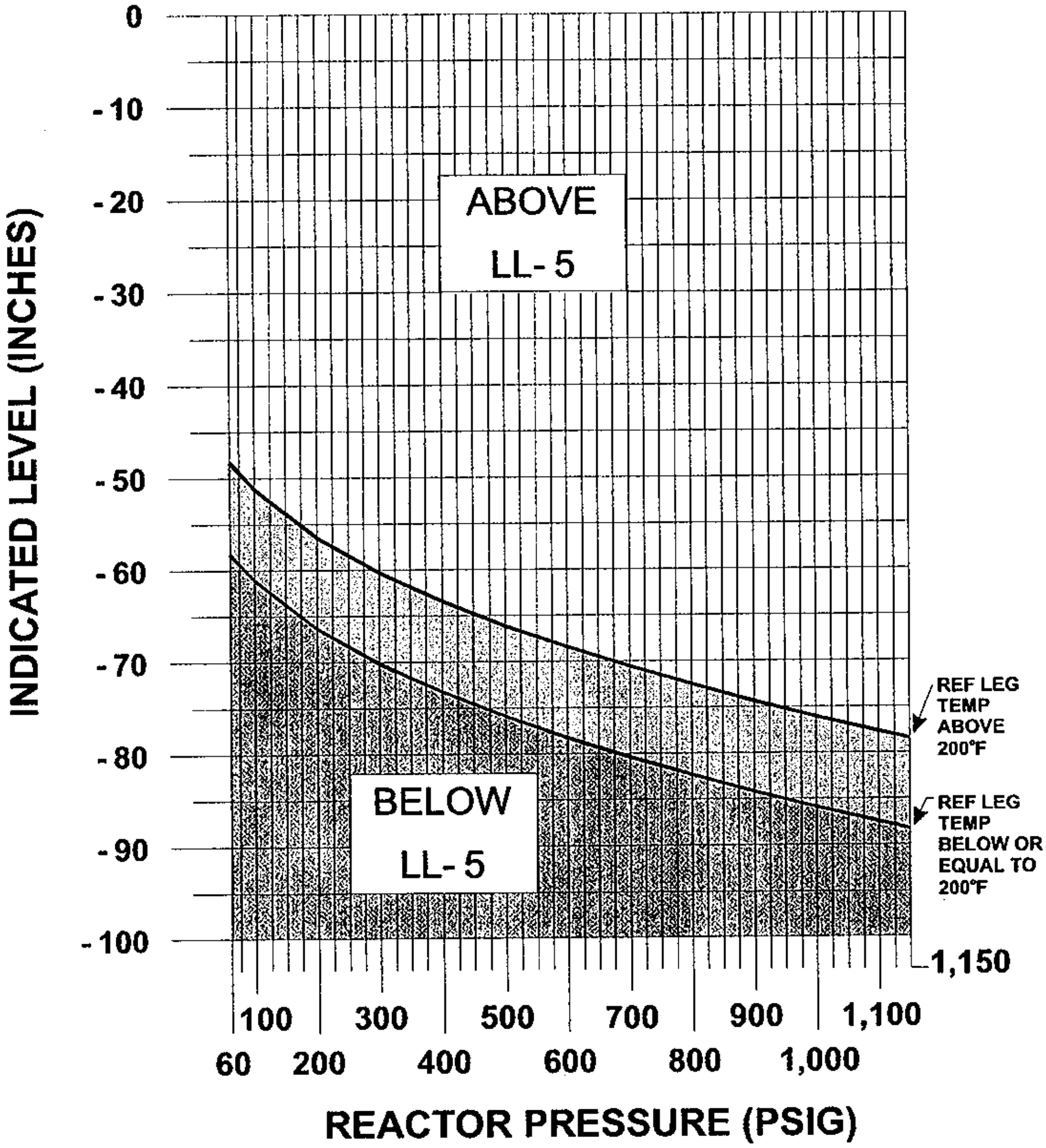


NOTE

WHEN REACTOR PRESSURE IS LESS THAN
60 PSIG, USE INDICATED LEVEL.
LL-4 IS -32.5 INCHES.

ATTACHMENT 6 (Cont'd)

FIGURE 19
UNIT 1 REACTOR WATER LEVEL AT LL-5
(MINIMUM ZERO INJECTION LEVEL)

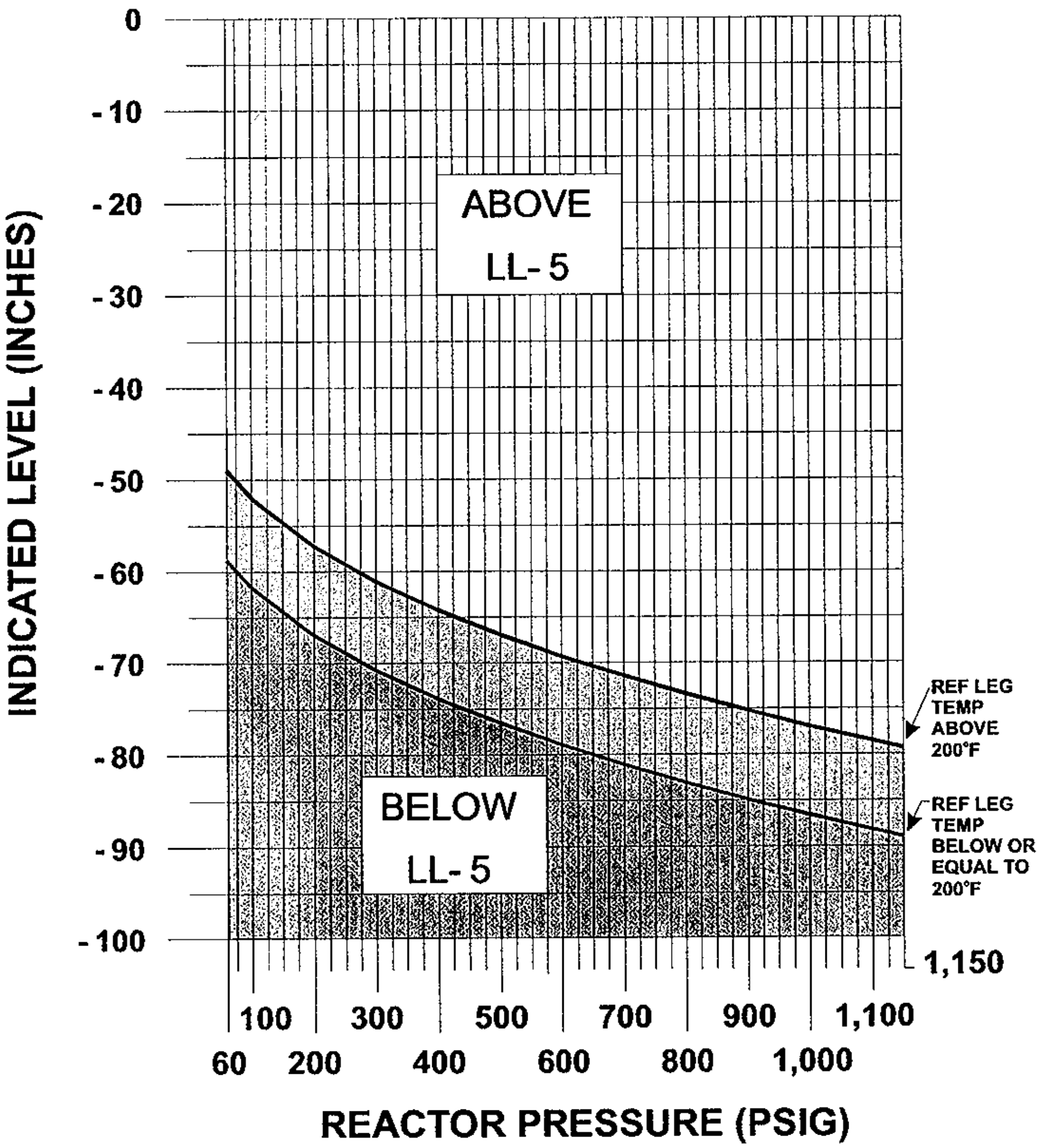


NOTE

WHEN REACTOR PRESSURE IS LESS THAN 60 PSIG, USE INDICATED LEVEL. LL-5 IS -47.5 INCHES.

ATTACHMENT 6 (Cont'd)

FIGURE 19A
UNIT 2 REACTOR WATER LEVEL AT LL-5
(MINIMUM ZERO INJECTION LEVEL)

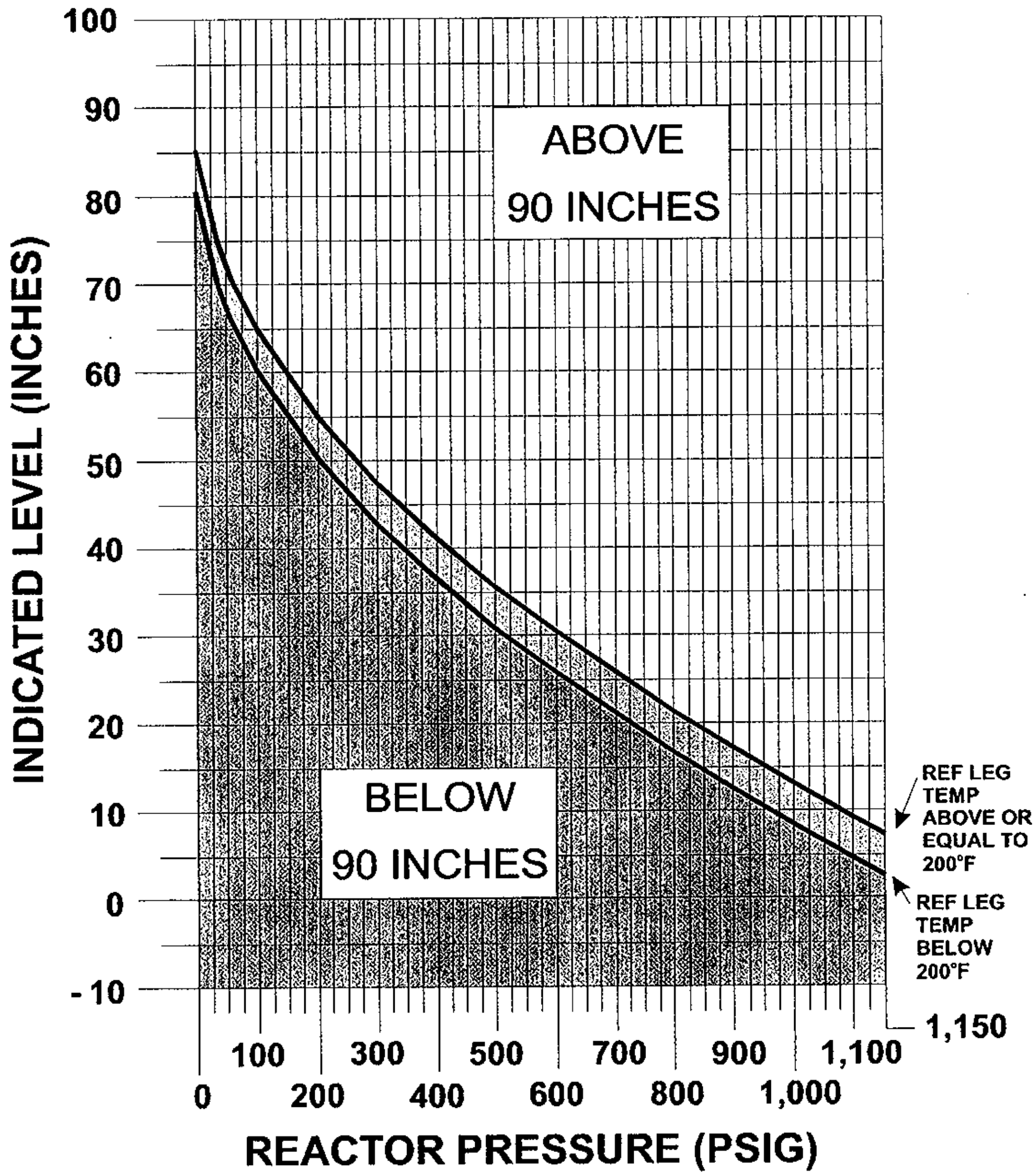


<u>NOTE</u>

WHEN REACTOR PRESSURE IS LESS THAN 60 PSIG, USE INDICATED LEVEL. LL-5 IS -47.5 INCHES.

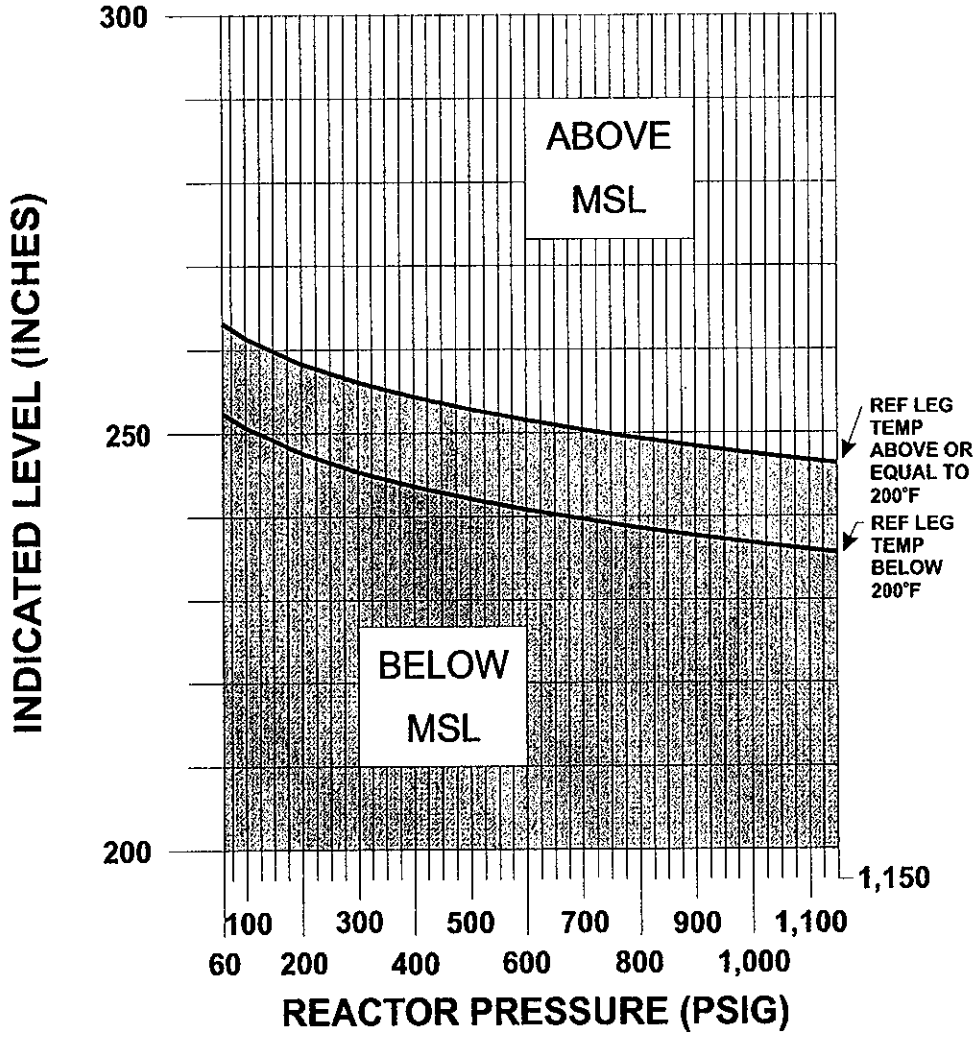
ATTACHMENT 6 (Cont'd)

FIGURE 20
REACTOR WATER LEVEL AT 90 INCHES



ATTACHMENT 6 (Cont'd)

FIGURE 21
REACTOR WATER LEVEL AT MSL
(MAIN STEAM LINE FLOOD LEVEL)



NOTE

WHEN REACTOR PRESSURE IS LESS THAN
60 PSIG, USE INDICATED LEVEL.
MSL IS +250 INCHES.