ATTACHMENT B

PROPOSED AMENDMENTS TO THE LICENSE/TECHNICAL SPECIFICATIONS

NPF-11	NPF-18
3/4 3-11	3/4 3-11
3/4 3-14*	3/4 3-14*
3/4 3-14(a)	3/4 3-14(a)
3/4 3-15	3/4 3-15
3/4 3-18	3/4 3-18
3/4 3-20	3/4 3-20
B3/4 3-2	B3/4 3-2

*This page is provided for information only, no changes.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
A. AUTOMATIC INITIATION				
1. PRIMARY CONTAINMENT ISOLATION.				
a. Reactor Vessel Water Level				
(1) Low, Level 3	7	2	1, 2, 3	20
(2) Low Low, Level 2	2, 3	2	1, 2, 3	20
(3) Low Low Low, Level 1	1, 10	2	1, 2, 3	20
b. Drywell Pressure - High	2, 7, 10	2	1, 2, 3	20
c. Main Steam Line				
1) Radiation - High	1	2	1, 2, 3	21
	3	2	1, 2, 3	22
2) Pressure - Low	1	2	1	23
3) Flow - High	1	2/line(d)	1, 2, 3	21
Deleted e d. Main Steam Line Tunnel Temperature - High	1	2	1(i)(j), 2(i)(j), 3(i)(j)	21 e
e. Main Steam Line Tunnel ΔTemperature - High	1	2	1(i)(j), 2(i)(j), 3(i)(j)	21
f. Condenser Vacuum - Low	1	2	1, 2*, 3*	21
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Vent Exhaust Plenum Radiation - High	4(c)(e)	2	1, 2, 3 and **	24
b. Drywell Pressure - High	4(c)(e)	2	1, 2, 3	24
c. Reactor Vessel Water Level - Low Low, Level 2	4(c)(e)	2	1, 2, 3, and #	24
d. Fuel Pool Vent Exhaust Radiation - High	4(c)(e)	2	1, 2, 3, and **	24

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTION

ACTION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 21 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 22 - Close the affected system isolation valves within 1 hour and declare the affected system inoperable.

ACTION 23 - Be in at least STARTUP within 6 hours..

ACTION 24 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within 1 hour.

ACTION 25 - Lock the affected system isolation valves closed within 1 hour and declare the affected system inoperable.

ACTION 26 - Provided that the manual initiation function is OPERABLE for each other group valve, inboard or outboard, as applicable, in each line, restore the manual initiation function to OPERABLE status within 24 hours; otherwise, restore the manual initiation function to OPERABLE status within 8 hours; otherwise:

a. Be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, or

b. Close the affected system isolation valves within the next hour and declare the affected system inoperable.

NOTES

* May be bypassed with reactor steam pressure $\leq$ 1043 psig and all turbine stop valves closed.

** When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

\# During CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

(a) Deleted.

(b) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the channel in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter. In addition for those trip systems with a design providing only one channel per trip system, the channel may be placed in an inoperable status for up to 8 hours for required surveillance testing without placing the channel in the tripped condition provided that the redundant isolation valve, inboard or outboard, as applicable, in each line is operable and all required actuation instrumentation for that redundant valve is OPERABLE, or place the trip system in the tripped condition.

(c) Also actuates the standby gas treatment system.

(d) A channel is OPERABLE if 2 of 4 instruments in that channel are OPERABLE.

(e) Also actuates secondary containment ventilation isolation dampers per Table 3.6.5.2-1.

(f) Closes only RWCU system inlet outboard valve.

This page is provided for information only, no changes

TABLE 3.3.2-1 (Continued)

NOTES (Continued)

(g) Requires RCIC steam supply pressure-low coincident with drywell pressure-high.

(h) Manual initiation isolates 1E51-F008 only and only with a coincident reactor vessel water level-low, level 2, signal.

(i) Both channels of each trip system may be placed in an inoperable status for up to 4 hours for required reactor building ventilation system corrective maintenance, filter changes, damper cycling and surveillance tests, other than Surveillance Requirement 4.6.5.1.c, without placing the trip system in the tripped condition.

(j) Both channels of each trip system may be placed in an inoperable status for up to 12 hours for performance of Surveillance Requirement 4.6.5.1.c without placing the trip system in the tripped condition.

due to loss of reactor building ventilation or

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
A. AUTOMATIC INITIATION		
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level		
1) Low, Level 3	≥ 12.5 inches*	≥ 11.0 inches*
2) Low Low, Level 2	≥ -50 inches*	≥ -57 inches*
3) Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches*
b. Drywell Pressure - High	≤ 1.69 psig	≤ 1.89 psig
c. Main Steam Line		
1) Radiation - High	≤ 3.0 x full power background	≤ 3.6 x full background
2) Pressure - Low	≥ 854 psig	≥ 834 psig
3) Flow - High	≤ 111 psid	≤ 116 psid
Deleted d. Main Steam Line Tunnel Temperature - High	≤ 140°F	≤ 146°F
e. Main Steam Line Tunnel Δ Temperature - High	≤ ~~36~~ 65 °F	≤ ~~42~~ 70 °F
f. Condenser Vacuum - Low	≥ 7 inches Hg vacuum	≥ 5.5 inches Hg vacuum
2. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Building Vent Exhaust Plenum Radiation - High	≤ 10 mr/hr	≤ 15 mr/hr
b. Drywell Pressure - High	≤ 1.69 psig	≤ 1.89 psig
c. Reactor Vessel Water Level - Low Low, Level 2	≥ -50 inches*	≥ -57 inches*
d. Fuel Pool Vent Exhaust Radiation - High	≤ 10 mr/hr	≤ 15 mr/hr
3. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. Δ Flow - High	≤ 70 gpm	≤ 87.5 gpm
b. Heat Exchanger Area Temperature - High	≤ 181°F	≤ 187°F
c. Heat Exchanger Area Ventilation ΔT - High	≤ 85°F	≤ 91°F
d. SLCS Initiation	NA	NA
e. Reactor Vessel Water Level - Low Low, Level 2	≥ -50 inches*	≥ -57 inches*

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
A. AUTOMATIC INITIATION	
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level	
1) Low, Level 3	N/A
2) Low Low, Level 2	N/A
3) Low Low Low, Level 1	≤ 1.0*
b. Drywell Pressure - High	N/A
c. Main Steam Line	
1) Radiation - High(**)	≤ 1.0*
2) Pressure - Low	≤ 2.0*
3) Flow - High	≤ 0.5*
Deleted d. Main Steam Line Tunnel Temperature - High	N/A e
e. Condenser Vacuum - Low	N/A
f. Main Steam Line Tunnel ΔTemperature - High	N/A
2. SECONDARY CONTAINMENT ISOLATION	N/A
a. Reactor Building Vent Exhaust Plenum Radiation - High	
b. Drywell Pressure - High	
c. Reactor Vessel Water Level - Low, Level 2	
d. Fuel Pool Vent Exhaust Radiation - High	
3. REACTOR WATER CLEANUP SYSTEM ISOLATION	N/A
a. ΔFlow - High	
b. Heat Exchanger Area Temperature - High	
c. Heat Exchanger Area Ventilation ΔT-High	
d. SLCS Initiation	
e. Reactor Vessel Water Level - Low Low, Level 2	
4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	N/A
a. RCIC Steam Line Flow - High	
b. RCIC Steam Supply Pressure - Low	
c. RCIC Turbine Exhaust Diaphragm Pressure - High	
d. RCIC Equipment Room Temperature - High	
e. RCIC Steam Line Tunnel Temperature - High	
f. RCIC Steam Line Tunnel ΔTemperature - High	
g. Drywell Pressure - High	
h. RCIC Equipment Room ΔTemperature - High	
5. RHR SYSTEM STEAM CONDENSING MODE ISOLATION	N/A
a. RHR Equipment Area ΔTemperature - High	
b. RHR Area Cooler Temperature - High	
c. RHR Heat Exchanger Steam Supply Flow High	

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
A. AUTOMATIC INITIATION				
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Level 3	S	Q	R	1, 2, 3
2) Low Low, Level 2	NA	Q	R	1, 2, 3
3) Low Low Low, Level 1	S	Q	R	1, 2, 3
b. Drywell Pressure - High	NA	Q	Q	1, 2, 3
c. Main Steam Line				
1) Radiation - High	S	Q	R	1, 2, 3
2) Pressure - Low	NA	Q	Q	1
3) Flow - High	NA	Q	R	1, 2, 3
Deleted				
d. Main Steam Line Tunnel Temperature - High	NA	Q	R	1, 2, 3
e. Condenser Vacuum - Low	NA	Q	Q	1, 2*, 3*
f. Main Steam Line Tunnel Δ Temperature - High	NA	Q	R	1, 2, 3
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Vent Exhaust Plenum Radiation - High	S	Q	R	1, 2, 3, and **
b. Drywell Pressure - High	NA	Q	Q	1, 2, 3
c. Reactor Vessel Water Level - Low Low, Level 2	NA	Q	R	1, 2, 3, and #
d. Fuel Pool Vent Exhaust Radiation - High	S	Q	R	1, 2, 3, and **
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	S	Q	R	1, 2, 3
b. Heat Exchanger Area Temperature - High	NA	Q	Q	1, 2, 3
c. Heat Exchanger Area Ventilation ΔT - High	NA	Q	Q	1, 2, 3
d. SLCS Initiation	NA	R	NA	1, 2, 3
e. Reactor Vessel Water Level - Low Low, Level 2	NA	Q	R	1, 2, 3

due to loss of reactor building ventilation or

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance. Both channels of each trip system for the main steam tunnel ambient temperature and ventilation system differential temperature may be placed in an inoperable status for up to 4 hours for required reactor building ventilation system maintenance and testing and 12 hours for the required secondary containment Leak Rate test without placing the trip system in the tripped condition. This will allow for maintaining the reliability of the ventilation system and secondary containment. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P-A, Supplement 2, "Technical Specification Improvement Analyses for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation", March 1989, and with NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation", July 1990. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains primary containment isolation capability. Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For A.C. operated valves, it is assumed that the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 13 seconds is assumed before the valve starts to move. The safety analysis considers an allowable inventory loss which in turn determines the valve speed in conjunction with the 13 second delay.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, trip setpoints and response times that will ensure effectiveness of the systems to provide the design protection. Although the instruments are listed by system, in some cases the same instrument may be used to send the actuation signal to more than one system at the same time.

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30936P-A, "Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation)", Parts 1 and 2, December 1988, and RE-025 Revision 1, "Technical Specification Improvement Analysis for the Emergency Core Cooling System Actuation Instrumentation for LaSalle County Station, Units 1 and 2", April 1991. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains ECCS initiation capability.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
A. AUTOMATIC INITIATION				
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
(1) Low, Level 3	7	2	1, 2, 3	20
(2) Low Low, Level 2	2, 3	2	1, 2, 3	20
(3) Low Low Low, Level 1	1, 10	2	1, 2, 3	20
b. Drywell Pressure - High	2, 7, 10	2	1, 2, 3	20
c. Main Steam Line				
1) Radiation - High	1	2	1, 2, 3	21
	3	2	1, 2, 3	22
2) Pressure - Low	1	2	1	23
3) Flow - High	1	2/line(d)	1, 2, 3	21
Deleted d. Main Steam Line Tunnel Temperature - High	1	2	1(i)(j), 2(i)(j), 3(i)(j)	21
e. Main Steam Line Tunnel ΔTemperature - High	1	2	1(i)(j), 2(i)(j), 3(i)(j)	21
f. Condenser Vacuum - Low	1	2	1, 2*, 3*	21
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Vent Exhaust Plenum Radiation - High	4(c)(e)	2	1, 2, 3 and **	24
b. Drywell Pressure - High	4(c)(e)	2	1, 2, 3	24
c. Reactor Vessel Water Level - Low Low, Level 2	4(c)(e)	2	1, 2, 3, and #	24
d. Fuel Pool Vent Exhaust Radiation - High	4(c)(e)	2	1, 2, 3, and **	24

ACTION STATEMENTS

ACTION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 21 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 22 - Close the affected system isolation valves within 1 hour and declare the affected system inoperable.

ACTION 23 - Be in at least STARTUP within 6 hours.

ACTION 24 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within 1 hour.

ACTION 25 - Lock the affected system isolation valves closed within 1 hour and declare the affected system inoperable.

ACTION 26 - Provided that the manual initiation function is OPERABLE for each other group valve, inboard or outboard, as applicable, in each line, restore the manual initiation function to OPERABLE status within 24 hours; otherwise, restore the manual initiation function to OPERABLE status within 8 hours; otherwise:

a. Be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, or

b. Close the affected system isolation valves within the next hour and declare the affected system in operable.

TABLE NOTATIONS

* May be bypassed with reactor steam pressure < 1043 psig and all turbine stop valves closed.

** When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

\# During CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

(a) Deleted.

(b) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the channel in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter. In addition for those trip systems with a design providing only one channel per trip system, the channel may be placed in an inoperable status for up to 8 hours for required surveillance testing without placing the channel in the tripped condition provided that the redundant isolation valve, inboard or outboard, as applicable, in each line is operable and all required actuation instrumentation for that redundant valve is OPERABLE, or place the trip system in the tripped condition.

(c) Also actuates the standby gas treatment system.

(d) A channel is OPERABLE if 2 of 4 instruments in that channel are OPERABLE.

(e) Also actuates secondary containment ventilation isolation dampers per Table 3.6.5.2-1.

(f) Closes only RWCU system inlet outboard valve.

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TABLE 3.3.2-1 (Continued)

NOTES (Continued)

(g) Requires RCIC steam supply pressure-low coincident with drywell pressure-high.

(h) Manual initiation isolates 2E51-F008 only and only with a coincident reactor vessel water level-low, level 2, signal.

(i) Both channels of each trip system may be placed in an inoperable status for up to 4 hours for required reactor building ventilation system corrective maintenance, filter changes, damper cycling and surveillance tests, other than Surveillance Requirement 4.6.5.1.c, without placing the trip system in the tripped condition.

(j) Both channels of each trip system may be placed in an inoperable status for up to 12 hours for performance of Surveillance Requirement 4.6.5.1.c without placing the trip system in the tripped condition.

due to loss of reactor building ventilation or

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
A. AUTOMATIC INITIATION		
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level		
1) Low, Level 3	≥ 12.5 inches*	≥ 11.0 inches*
2) Low Low, Level 2	≥ -50 inches*	≥ -57 inches*
3) Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches*
b. Drywell Pressure - High	≤ 1.69 psig	≤ 1.89 psig
c. Main Steam Line		
1) Radiation - High	≤ 3.0 x full power background	≤ 3.6 x full background
2) Pressure - Low	≥ 854 psig	≥ 834 psig
3) Flow - High	≤ 111 psid	≤ 116 psid
Deleted d. Main Steam Line Tunnel Temperature - High e	≤ 140°F e	≤ 146°F e
e. Main Steam Line Tunnel Δ Temperature - High	≤ ~~36°F~~ 65	≤ ~~42°F~~ 70
f. Condenser Vacuum - Low	≥ 7 inches Hg vacuum	≥ 5.5 inches Hg vacuum
2. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Building Vent Exhaust Plenum Radiation - High	≤ 10 mr/h	≤ 15 mr/h
b. Drywell Pressure - High	≤ 1.69 psig	≤ 1.89 psig
c. Reactor Vessel Water Level - Low Low, Level 2	≥ -50 inches*	≥ -57 inches*
d. Fuel Pool Vent Exhaust Radiation - High	≤ 10 mr/h	≤ 15 mr/h
3. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. ΔFlow - High	≤ 70 gpm	≤ 87.5 gpm
b. Heat Exchanger Area Temperature - High	≤ 181°F	≤ 187°F
c. Heat Exchanger Area Ventilation ΔT - High	≤ 85°	≤ 91°F
d. SLCS Initiation	N.A.	N.A.
e. Reactor Vessel Water Level - Low Low, Level 2	≥ -50 inches*	≥ -57 inches*

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
A. AUTOMATIC INITIATION	
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level	
1) Low, Level 3	N/A
2) Low Low, Level 2	N/A
3) Low Low Low, Level 1	≤ 1.0*
b. Drywell Pressure - High	N/A
c. Main Steam Line	
1) Radiation - High(**)	≤ 1.0*
2) Pressure - Low	≤ 2.0*
3) Flow - High	≤ 0.5*
d. Main Steam Line Tunnel Temperature - High	N/A
e. Condenser Vacuum - Low	N/A
f. Main Steam Line Tunnel ΔTemperature - High	N/A
2. SECONDARY CONTAINMENT ISOLATION	N/A
a. Reactor Building Vent Exhaust Plenum Radiation - High	
b. Drywell Pressure - High	
c. Reactor Vessel Water Level - Low, Level 2	
d. Fuel Pool Vent Exhaust Radiation - High	
3. REACTOR WATER CLEANUP SYSTEM ISOLATION	N/A
a. ΔFlow - High	
b. Heat Exchanger Area Temperature - High	
c. Heat Exchanger Area Ventilation ΔT-High	
d. SLCS Initiation	
e. Reactor Vessel Water Level - Low Low, Level 2	
4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	N/A
a. RCIC Steam Line Flow - High	
b. RCIC Steam Supply Pressure - Low	
c. RCIC Turbine Exhaust Diaphragm Pressure - High	
d. RCIC Equipment Room Temperature - High	
e. RCIC Steam Line Tunnel Temperature - High	
f. RCIC Steam Line Tunnel ΔTemperature - High	
g. Drywell Pressure - High	
h. RCIC Equipment Room ΔTemperature - High	
5. RHR SYSTEM STEAM CONDENSING MODE ISOLATION	N/A
a. RHR Equipment Area ΔTemperature - High	
b. RHR Area Cooler Temperature - High	
c. RHR Heat Exchanger Steam Supply Flow High	

Deleted

e

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
A. AUTOMATIC INITIATION				
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Level 3	S	Q	R	1, 2, 3
2) Low Low, Level 2	NA	Q	R	1, 2, 3
3) Low Low Low, Level 1	S	Q	R	1, 2, 3
b. Drywell Pressure - High	NA	Q	Q	1, 2, 3
c. Main Steam Line				
1) Radiation - High	S	Q	R	1, 2, 3
2) Pressure - Low	NA	Q	Q	1
3) Flow - High	NA	Q	R	1, 2, 3
d. Main Steam Line Tunnel Temperature - High	NA	Q	R	1, 2, 3
e. Condenser Vacuum - Low	NA	Q	Q	1, 2*, 3*
f. Main Steam Line Tunnel Δ Temperature - High	NA	Q	R	1, 2, 3
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Vent Exhaust Plenum Radiation - High	S	Q	R	1, 2, 3 and **
b. Drywell Pressure - High	NA	Q	Q	1, 2, 3
c. Reactor Vessel Water Level - Low Low, Level 2	NA	Q	R	1, 2, 3, and #
d. Fuel Pool Vent Exhaust Radiation - High	S	Q	R	1, 2, 3 and **
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	S	Q	R	1, 2, 3
b. Heat Exchanger Area Temperature - High	NA	Q	Q	1, 2, 3
c. Heat Exchanger Area Ventilation ΔT - High	NA	Q	Q	1, 2, 3
d. SLCS Initiation	NA	R	NA	1, 2, 3
e. Reactor Vessel Water Level - Low Low, Level 2	NA	Q	R	1, 2, 3

Deleted

due to loss of reactor building ventilation or

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance. Both channels of each trip system for the main steam tunnel ambient temperature and ventilation system differential temperature may be placed in an inoperable status for up to 4 hours for required reactor building ventilation system maintenance and testing and 12 hours for the required secondary containment Leak Rate test without placing the trip system in the tripped condition. This will allow for maintaining the reliability of the ventilation system and secondary containment. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P-A, Supplement 2, "Technical Specification Improvement Analyses for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation", March 1989, and with NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation", July 1990. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains primary containment isolation capability. Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For A.C. operated valves, it is assumed that the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 13 seconds is assumed before the valve starts to move. The safety analysis considers an allowable inventory loss which in turn determines the valve speed in conjunction with the 13 second delay.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, trip setpoints and response times that will ensure effectiveness of the systems to provide the design protection. Although the instruments are listed by system, in some cases the same instrument may be used to send the actuation signal to more than one system at the same time.

ATTACHMENT C

SIGNIFICANT HAZARDS CONSIDERATION

Commonwealth Edison has evaluated the proposed Technical Specification Amendment and determined that it does not represent a significant hazards consideration. Based on the criteria for defining a significant hazards consideration established in 10 CFR 50.92, operation of LaSalle County Station Units 1 and 2 in accordance with the proposed amendment will not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated because:

   a. There is no effect on accident initiators so there is no change in probability of an accident. The accident analysis associated with a steam line break in the main steam line tunnel assumes an instantaneous circumferential break of a main steam line downstream of the outermost isolation valve. The leak detection isolation on differential temperature based on ≤ 10% of a calculated critical crack of a main steam line is only a precursor of a break, and thus does not affect the probability of a break.

   b. There is no or minimal effect on the consequences of analyzed accidents due to deletion of the automatic isolation on high temperature leak detection in the main steam line tunnel or due to increasing the leak detection differential temperature setpoint and allowable values to detect a 100 gpm steam leak from a crack in a main steam line. The worst case accident corresponding to main steam lines outside of the reactor vessel and primary containment boundary is a main steam line break, which bounds the dose consequences of any size steam leak less than a full break. Also, a 200 gpm steam leak results in a calculated offsite dose within the annual whole body dose limit and the radioiodine release limit per 10CFR50 Appendix I, if detected and isolated within several weeks.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated because:

   The purpose of the main steam line isolation is based on leak detection and automatic isolation for leakage in the main steam line tunnel downstream of the outermost isolation valve. This change maintains this capability with only the leak detection based on high differential temperature in the steam line tunnel. Also, the primary containment isolation logic for main steam line leak detection isolation on high differential temperature remains the same.Thus no new or different accident is created.

ATTACHMENT C

SIGNIFICANT HAZARDS CONSIDERATION

3) Involve a significant reduction in the margin of safety because:

The increased setpoint for differential temperature leak detection for automatic isolation of the main steam lines due to a steam leak outside of the primary containment is based on calculated/analyzed response to a steam leak small compared to the leak from a critical crack. The leak detection isolation logic remains single failure proof. The previous evaluation of diversity of isolation parameters considered the ambient temperature and differential temperature isolations as one parameter in Table 5.2-8 of the LaSalle UFSAR. The deletion of leak detection isolation of the main steam lines based on high ambient temperature in the main steam line tunnel is acceptable, because the differential temperature isolation has been analyzed to detect and isolate the main steam lines based on bounding inlet air temperatures. Therefore, the Main Steam Line High flow, vessel low level, and the differential temperature instruments maintain adequate diversity of isolation parameters without main steam line tunnel high temperature.

The differential temperature leak detection for the main steam line tunnel depends on normal ventilation flow to detect leakage. Therefore, the trip function will be declared inoperable upon loss of or shutdown of normal ventilation. The Technical Specifications currently allow the main steam tunnel high temperature and high differential temperature isolation channels to be inoperable for up to 4 or 12 hours during the performance of specified required surveillances. The 12 hours allowed outage time is currently for an 18 month surveillance requirement. The addition of allowance for up to 12 hours allowed outage time to recover normal ventilation following an unplanned loss of normal ventilation is reasonable, since the time is small compared to the time frame over which a pipe crack grows. Also, supplemental monitoring of water collection sumps and area temperature in the main steam line tunnel provides heightened awareness of operators to detect leakage in the main steam line tunnel during the time normal ventilation is not available. The planned shutdown of normal ventilation is currently allowed for up to 4 hours by the Technical Specifications. The unplanned loss of normal ventilation is expected to be less than two times per cycle upon completion design changes to make the isolation logic power supply D.C. instead of A.C. through motor generator sets.

Therefore, this requested Technical Specification amendment does not involve a significant reduction in the margin of safety.

ATTACHMENT C

SIGNIFICANT HAZARDS CONSIDERATION

Guidance has been provided in "Final Procedures and Standards on No Significant Hazards Considerations," Final Rule, 51 FR 7744, for the application of standards to license change requests for determination of the existence of significant hazards considerations. This document provides examples of amendments which are and are not considered likely to involve significant hazards considerations. These proposed amendments most closely fit the example of a change which may either result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the applicable Standard Review Plan.

This proposed amendment does not involve a significant relaxation of the criteria used to establish safety limits, a significant relaxation of the bases for the limiting safety system settings or a significant relaxation of the bases for the limiting conditions for operations. Therefore, based on the guidance provided in the Federal Register and the criteria established in 10 CFR 50.92(c), the proposed change does not constitute a significant hazards consideration.

ATTACHMENT D

ENVIRONMENTAL ASSESSMENT STATEMENT APPLICABILITY REVIEW

Commonwealth Edison has evaluated the proposed amendment against the criteria for identification of licensing and regulatory action requiring environmental assessment in accordance with 10 CFR 51.21. It has been determined that the proposed changes meet the criteria for a categorical exclusion as provided under 10 CFR 51.22(c)(9). This conclusion has been determined because the changes requested do not pose significant hazards considerations or do not involve a significant increase in the amounts, and no significant changes in the types, of any effluents that may be released off-site. Additionally, this request does not involve a significant increase in individual or cumulative occupational radiation exposure.

ATTACHMENT E

UFSAR REFERENCES

1. Table 7.1-8 (7.1-9 in FSAR), Leak Detection System Codes and Standards.

2. PAGE 7.A.3-35 OF SECTION 7.A.3.2.1. CONFORMANCE TO IEEE 279-1971. OPERATING BYPASSES, IEEE 279-1971 Paragraph 4.8. (FSAR section 7.3.2.2.1)

3. PAGE 7.A.3-34 OF SECTION 7.A.3.2.1, CONFORMANCE TO IEEE 279-1971. DERIVATION OF SYSTEM INPUTS, IEEE 279-1971 Paragraph 4.8. (FSAR section 7.3.2.2.1)

4. 7.3.2.2.3.3 Safety Design Bases of the initiating isolation signal, Main Steamline Space High Temperature and Differential Temperature. (FSAR section 7.3.1.1.2.4.3)

5. BTP ICSB 21 Guidance for Application of Regulatory Guide 1.47, STP pp. 7.A-15, ; and R.G. 1.47, Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems.

ATTACHMENT F

HISTORY OF LASALLE UNIT 1 AND 2 MAIN STEAM TUNNEL LEAK DETECTION ISOLATIONS

From the initial startup of LaSalle Unit 1, it was recognized that the setpoints of the high differential temperature leak detection for Main Steam Line isolation were very close to the normal operating conditions. This was not a significant problem during steady state conditions, but during transient conditions related to changes in main steam tunnel ventilation (VR) flow and/or inlet temperature, the instruments falsely detected a leak. Flow changes which cause the Main Steam Tunnel (MST) temperature to approach the high temperature trip have been caused by isolation of VR (no flow) or filter plugging (which has occurred quickly due to blizzards). A significant drop in inlet temperature which causes MST differential temperature to approach the differential temperature trip can be caused by loss of the Station Heat Recovery system (SH), loss of the electric heaters or even sudden drops in outside air temperature. Over the years many attempts have been made to mitigate this problem. Spurious trips of these instruments result in a scram which isolates the reactor from its primary heat sink causing a significant challenge to the plant and its operators.

The high temperature channels did not originally cause much of a problem due to their location in the steam tunnel. VR could be shutdown for several hours without tripping the high temperature sensors. The differential temperature sensors were not a problem with VR shutdown (differential temperature goes more slowly), but would trip when VR was restarted due to an inrush of cooler air prior to the outlet sensors cooling off or when running the purge train (VQ) with the suction off the steam tunnel. Solutions at that time provided procedural guidance for methods to avoid the trip of the differential temperature sensors when VR was shutdown and started. The initial solution to this problem was a Technical Specification change to allow the differential temperature channels to be bypassed for up to four hours if the temperature channels were operable. This Technical Specification allowance was provided in the initial Unit 2 Technical Specification and Unit 1 Technical Specification amendment 18 (August 1984). On May 25, 1984 an exigent Technical Specification change was submitted (OSR 84-25) to change the differential temperature setpoints based on actual steady state differential temperature data, keeping the calculational basis (25 gpm) unchanged. The original Technical Specification setpoint was based on expected conditions in the main steam tunnel, however the actual differential temperature was greater than pre-startup calculations predicted. This amendment (Unit 1 #17, Unit 2 #2) was approved. This provided more operating margin for steady state conditions, but did not prevent isolations from occurring if VR was restarted. The allowed bypass of the differential temperature sensors provided the allowance to restart VR without a trip by installing jumpers.

ATTACHMENT F

HISTORY OF LASALLE UNIT 1 AND 2 MAIN STEAM TUNNEL LEAK DETECTION ISOLATIONS

However to make the Leak Detection system more responsive due to an overall Leak Detection review, modifications were initiated to move the temperature and differential temperature sensors in both units. After the Unit 1 sensors were moved, on February 2, 1985, the unit scrammed due to high temperature soon after VR was isolated. The installation of the modification was delayed in Unit 2 based on this problem and is still not installed. Since the temperature sensors were relocated, the temperature channels have been a problem in Unit 1.

As a result of the isolation an engineering study was initiated to determine alternatives to the temperature and differential temperature Leak Detection design. The study concluded that there were technically viable designs which could be pursued to partially mitigate the problem. The engineering solutions were expensive and were not total solutions to the problem. In parallel a PRA evaluation with EPRI help was performed to confirm from a risk basis that a small steam leak did not warrant a MSIV closure transient. Another Technical Specification change was pursued on July 10, 1987 (OSR 87-27). This amendment request proposed to delete the temperature and differential temperature isolation signals as a safety improvement. However, the submittal was incomplete and was rejected on February 8, 1988.

The major concern at the time was performing VR isolation damper surveillances and the secondary containment leakage rate test (SCLRT). Up to this point, all SCLRTs were done during dual unit outages since VR must be shut down in both units to perform the test. Due to the urgent need of Technical Specification relief to avoid a dual unit outage, another submittal was made (OSR 89-41) on July 26, 1989 to bypass the trips for a limited time to accomplish the SCLRT and other expected routine activities. This change allowed the station to perform maintenance work and surveillance testing which was the primary concern in the short term and allowed the amendment to be approved in a relatively short time (Unit 1 # 77, Unit 2 # 61).

On August 16, 1995 Unit 1 secondary containment isolated due to a loss of power to the B RPS bus. Seven minutes after the loss of ventilation, the Unit 1 steam tunnel temperature reached the setpoint of 140 °F and caused the isolation of the main steam lines. The MSIV isolation caused the reactor to scram. It was obvious the entire problem had not been resolved.

Over the weekend of September 9 and 10, 1995, the Unit 2 differential temperature on one channel monitoring the upper steam tunnel reached 34 °F and another channel reached 32 °F. The trip setpoint is 36 °F. This has occurred as a result of cool nights, and not steam or water leaks.

As a result of non-leak conditions as indicated above there is little margin to a MSIV closure and associated automatic scram due to a false leak detection signal. As discussed in the history above, the steam tunnel temperatures and differential temperatures are affected by many things, which challenges the operators and at times places the unit(s) in jeopardy. The need to evaluate changing the setpoints or eliminate automatic isolation on temperature and differential temperature was reinforced as a result of the Unit 1 scram August 16, 1995. The urgency of this evaluation was amplified after the high differential temperatures on Unit 2 on September 9 and 10, 1995. Since then, many calculations and associated analyses have been required in order to determine a new basis for leakage detection isolation for the main steam lines outside of the primary containment isolation boundary. LaSalle Unit 1 is currently scheduled to begin shutdown for its seventh refuel outage, L1R07, on January 25, 1995. Due to the short time allowed for operator action to install jumpers to bypass the Main Steam Line Tunnel Temperature - High main steam line isolation automatic isolation, this "operator work around" has been determined to be the most significant in NRC Region 3. LaSalle has initiated design changes in parallel with the proposed Technical Specification amendment request. The Unit 1 design changes are being scheduled to be installed during L1R07.

ATTACHMENT G

CALCULATION SUMMARIES

1. Critical Crack Calculation and Steam Leakage Flow Calculation Summary:

The ComEd calculation NED-P-MSD-086, "LaSalle Station Steam Tunnel Main Steam Line Leak Rate Calculation" Rev.0, dated December 5, 1995, determines the critical crack size and its projected leakage for the Main Steam lines downstream of the outboard isolation valves in the MST. These quantities are calculated to determine the maximum possible leak rate, prior to line rupture for the Main Steam lines during normal operation. The calculation determines the critical crack size for a circumferentially oriented throughwall flaw using a limit load approach. The limit load methodology defined in the ASME Boiler and Pressure Vessel Code, Section XI, 1989 Edition, is used as guidance for this calculation. This methodology has also been validated by GE for this application in LaSalle Leak Detection Temperature Measurement", GENE Report DRF-E31-00029-3C, November 1995. To calculate the leak rate from the critical crack, the PICEP, Pipe Crack Evaluation Program, EPRI NP-3596-SR, December 1987, is used.

The results of this calculation demonstrate the critical crack length to be 33.2" long. The leak rate for a circumferential flaw of this length is approximately 1290 gpm. It is the station's intent to use the minimum leakage that will ensure adequate operational margin. The value being considered but not finalized is 100 gpm. This represents less than 10% of the critical crack length leakage.

2. Calculation of changes in differential temperature in response to a steam leak:

ComEd calculation, LaSalle Main Steam Tunnel Temperature Response due to Steam Leakage with Ventilation System in Operation, BSA-L-95-05, Rev. 0, dated January 13, 1996. This calculation uses the GOTHIC computer code to determine the main steam tunnel temperature (MST) response due to a variety of leak rates and supply air temperatures. GOTHIC (Generation Of Thermal-Hydraulic Information for Containments) is a general purpose thermal-hydraulics computer program for design, licensing, safety and operating analysis of nuclear power plant containments and other confinement buildings. Applications of GOTHIC include evaluations of containment and containment subcompartment response to the full spectrum of high energy line breaks within the design basis envelope as described in UFSAR chapter 6, section 2.

ATTACHMENT G

CALCULATION SUMMARIES

Applications may include pressure and temperature determination, equipment qualification profiles and inadvertent system initiation, and degradation or failure of engineered safety features.

A sketch of the main steam tunnel is shown in Figure 1. In this analysis the MST is subdivided into three areas denoted as the lower, middle, and upper MST. The lower and upper MST extend from elevation 687 to 706 ft and from elevation 736 ft 7 in to 768 ft 4.5 in, respectively. The middle MST consists of the vertical section of the tunnel between elevation 706 ft and 736 ft 7 in.

The temperature of the MST is maintained by the reactor building ventilation (VR) system. Two streams of VR air enter the MST, one at 687' elevation and the other at 740' elevation and exit the tunnel via an exhaust riser located at the top of the MST.

The analysis assumes the crack and steam leak develop over a 100 second period in the main steam line. The cracks are postulated to occur at any location along the full length of the steam line in the MST. For leak detection purposes, the lowest temperature rise in the MST would form the basis for establishing a leak detection setpoint. The lowest temperature rise in the MST, due to a given amount of steam leakage, occurs near the exit of the upper MST VR flow and with a crack located in the lower MST near the area where the steam lines exit the tunnel. Thus, in this analysis, only a leak in the lower MST was considered.

The analytical model consists of three control volumes representing the upper, middle and lower MST. Five additional control volumes are used to model the structures adjacent to the MST. A flow path is used to represent the flow from the lower to the middle MST and another path from the middle to the upper MST. Additional flow paths are used to model the VR inlet and exhaust flows. The heat load in the MST is produced predominantly from the main steam and feedwater lines and is represented by one heater located in each of the MST control volumes. The VR supply air originates from outside the plant and enters the MST after passing through various areas in the reactor building. Hence, the VR supply air temperature is dependent on the outdoor air temperature. Two different supply air temperature values, 65 and 110 °F, are used to bound the conditions expected year round. The VR flow rates are based on plant data and are:

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CALCULATION SUMMARIES

Upper MST supply flow	24,000 cfm
Lower MST supply flow	40,000 cfm
Exhaust MST flow	64,000 cfm

The upper MST supply flow and the exhaust MST flow are held constant during the analysis. The lower MST supply flow is varied in relation to the steam leakage rate according to a mass balance in and out of the MST.

The GOTHIC computer code results were compared with plant data and show good agreement between the calculated results and plant data.

The results show that the upper MST temperature varies directly with the quantity of steam leakage. For a given leakage rate, the upper MST temperature is highest for a VR inlet temperature of 110 °F and lowest for a VR inlet temperature of 65 °F.

The data (see Figure 2 attached) indicates that with a 65 °F VR inlet temperature and a 125 gpm leak in the lower MST, the upper MST temperature would reach a value of approximately 164 °F. Using a leak detection setpoint for the temperature high of 164 °F would still result in the requirement to quickly bypass the isolation logic for a loss of VR in order to avoid the main steam isolation. The required response of bypassing the logic becomes even more critical on hotter days. In order to eliminate the need to bypass the leak detection logic on a loss of VR, a temperature high setpoint of ~200 °F would be required. The crack leakage required to generate a 200 °F temperature in the upper MST with VR in operation with a 65 °F inlet air temperature would be several hundred gpm. Therefore, leak detection isolation based on Main Steam Line Tunnel Temperature High would not allow operation without the potential for spurious trips.

The results also show that the upper MST differential temperature (the temperature difference between the exhaust riser air temperature and the VR inlet air temperature) varies directly with the quantity of steam leakage. For a given leakage rate, the upper MST differential temperature is highest for a VR inlet temperature of ≤ 65 °F and lowest for a VR inlet temperature of ≥ 110 °F.

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CALCULATION SUMMARIES

The data (see Figure 3, attached) indicates that with a 110 °F VR inlet temperature and a 100 gpm leak in the lower MST, the upper MST differential temperature would reach a value of 72.5 °F. It is estimated that over 99% of the year the VR inlet temperature will be less than or equal to 100 °F. With the VR inlet temperature less than 110 °F, the setpoint will actually be isolating the main steam lines on a leak of less than 100 gpm.

The current MST leak detection temperature instruments would remain available for alarm and indication capability. With a loss of VR, the differential temperature instrumentation would be made technically inoperable. The current temperature instruments would then be used to augment a monitoring surveillance of the MST to confirm no leaks develop while the differential temperature instrumentation is inoperable. If a leak is confirmed, manual isolation could be initiated as required.

Because the VR flows are induced by the VR exhaust fans, the inlet VR flow is reduced by the presence of steam. The reduce inlet flow phenomenon may affect the temperature indication from a sensor located in the VR inlet. As the air inlet flow reduces to zero due to increased leakage flow, the temperature indicated will be that of the MST instead of the inlet flow. An analysis was performed that determined that both upper and lower VR flows are reduced to zero for a 250 gpm steam leak that grows from zero to 250 gpm over a 100 second period. Isolating on a 100 gpm steam leak assures that leak detection on differential temperature remains operable.

3. Setpoint Calculation:

Main Steam Tunnel Temperature Isolation Setpoint Error Analysis. Calculation No. NED-I-EIC-0208, Dated January 16, 1996.

The purpose of this calculation is to determine the calibrated setpoint and allowable value for the Main Steam Tunnel Area Vent High Differential Temperature Isolation instrument channels. This channel provides a Technical Specification setpoint and initiates a Group I isolation of the MSIVs and MSL drain valves on high differential temperature.

The calculation is valid under normal operating, accident environmental conditions, and allows for all normal operating and accident errors, thus ensuring Technical Specification compliance for the instrument channels.

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CALCULATION SUMMARIES

The methodology used herein is based on ComEd documents,

1. ComEd TID-E/I&C-20, "Basis for Analysis of Instrument Channel Set-point Error & Loop Accuracy", Rev. 0, dated April 6, 1992, and

2. ComEd TID-E/I&C-10, "Analysis of Instrument Channel Setpoint Error and Instrument Loop Accuracy", Rev. 0, dated April 6, 1992.

The primary basis for ComEd setpoint methodology is ANSI/ISA-S67.04-1988, "Setpoints for Nuclear Safety Related Instrumentation."

The evaluation of errors used to determine the "Total Error" (TE) is consistent with the above methodology with the following exception:

> The calibration tolerance is assumed to describe the limits of the as-left component outputs. For a random error, this corresponds to 100% of the population and can be statistically represented by a 3 sigma value. Per the above TIDs , the "Setting Tolerance" (ST) is defined as a random error which is due to the procedural allowances given to the technician performing the calibration.

As stated above, the objective of this calculation was to determine the available margin between the Tech Spec allowable value and the nominal trip setpoint for normal operating conditions and the margin between the analytical limit and the nominal trip setpoint for accident conditions. The acceptance criteria for this calculation was that a positive margin is required.

Conclusions:

The calculation determined a nominal setpoint for the main steam line tunnel differential temperature isolation setpoint that ensures a high level of confidence that the analytical limit will not be exceeded under normal or accident operating conditions.

1. The nominal trip setpoint is determined to be $\leq$ 65.6 °F.

2. The allowable value is determined to be $\leq$ 70.1 °F.

ATTACHMENT G

CALCULATION SUMMARIES

4. 10CFR20 calculation of the offsite dose consequences due to a 200 gpm steam leak in the main steam line tunnel:

The off-site radiological doses due to postulated steam leakage in the LaSalle County Station main steam line tunnel were calculated based on the normal operating condition radionuclide source documented in General Electric Report NEDO 10871, dated March 1973, "Technical Derivation of BWR 1971 Design Basis Radioactive Material Source Terms". This corresponds to "failed fuel" releasing 100,000 microcuries of noble gases per second after 30 minutes delay along with 700 microcuries per second of radioiodines. The off-site dose assessment methodology and site specific data are included in the ComEd Offsite Dose Calculation Manual (ODCM).

The calculation, Radiological Doses Due to Postulated Steam Tunnel Leakage, BSA-L-96-03, Rev. 0, dated 1/12/96, indicates that a steam leak of 200 gallons per minute will result in a whole body dose rate of 1.9E-3 mrads per hour. The 10CFR50, Appendix I whole body annual dose limit is 5 mrem per year. At 1.9E-3 mrads per hour, it take over 2600 hours to exceed the 5 mrem limit. The radioiodine release rate is 9.8E-2 microcuries per second, and for a short duration is small compared to the limit of 1 curie per year in Appendix I.

In conclusion, a steam leak of 200 gpm could continue for several weeks without exceeding Appendix I limits. Thus, isolation of a main steam line on a steam leak of 100 gpm is well below the normal operation radionuclide release limits of 10CFR50, Appendix I.

FIGURE 1
MAIN STEAM LINE TUNNEL

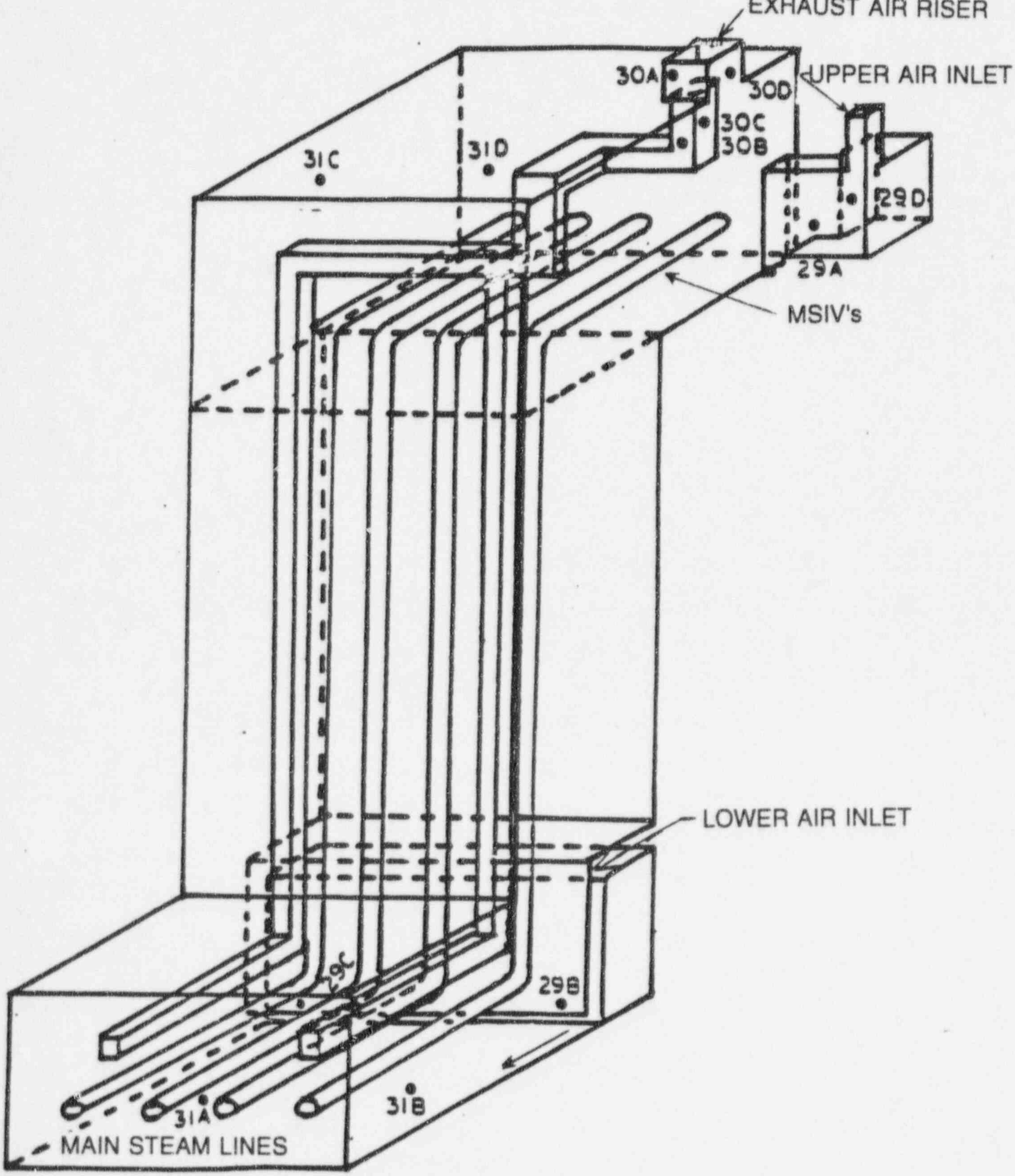


Figure 2

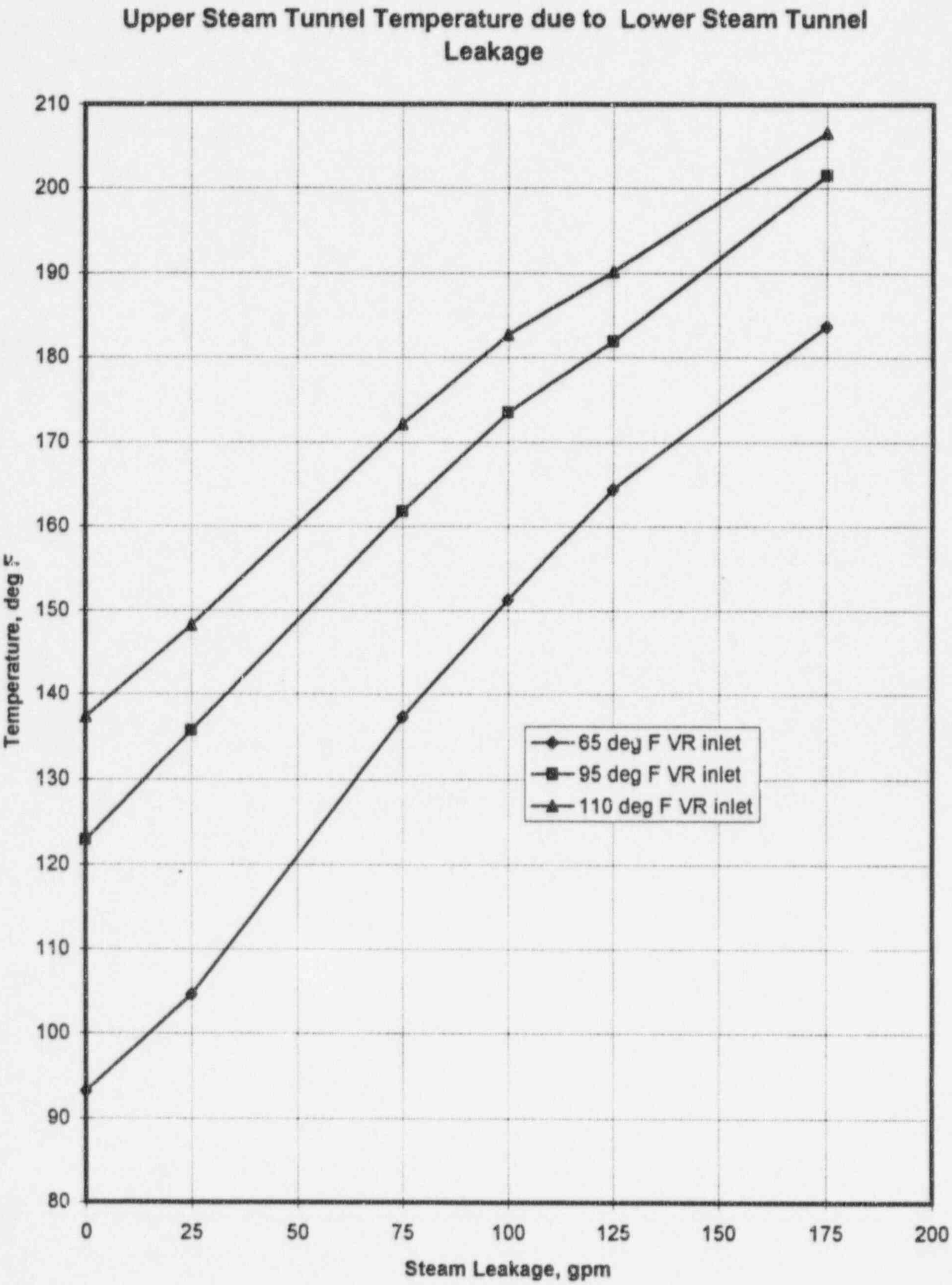
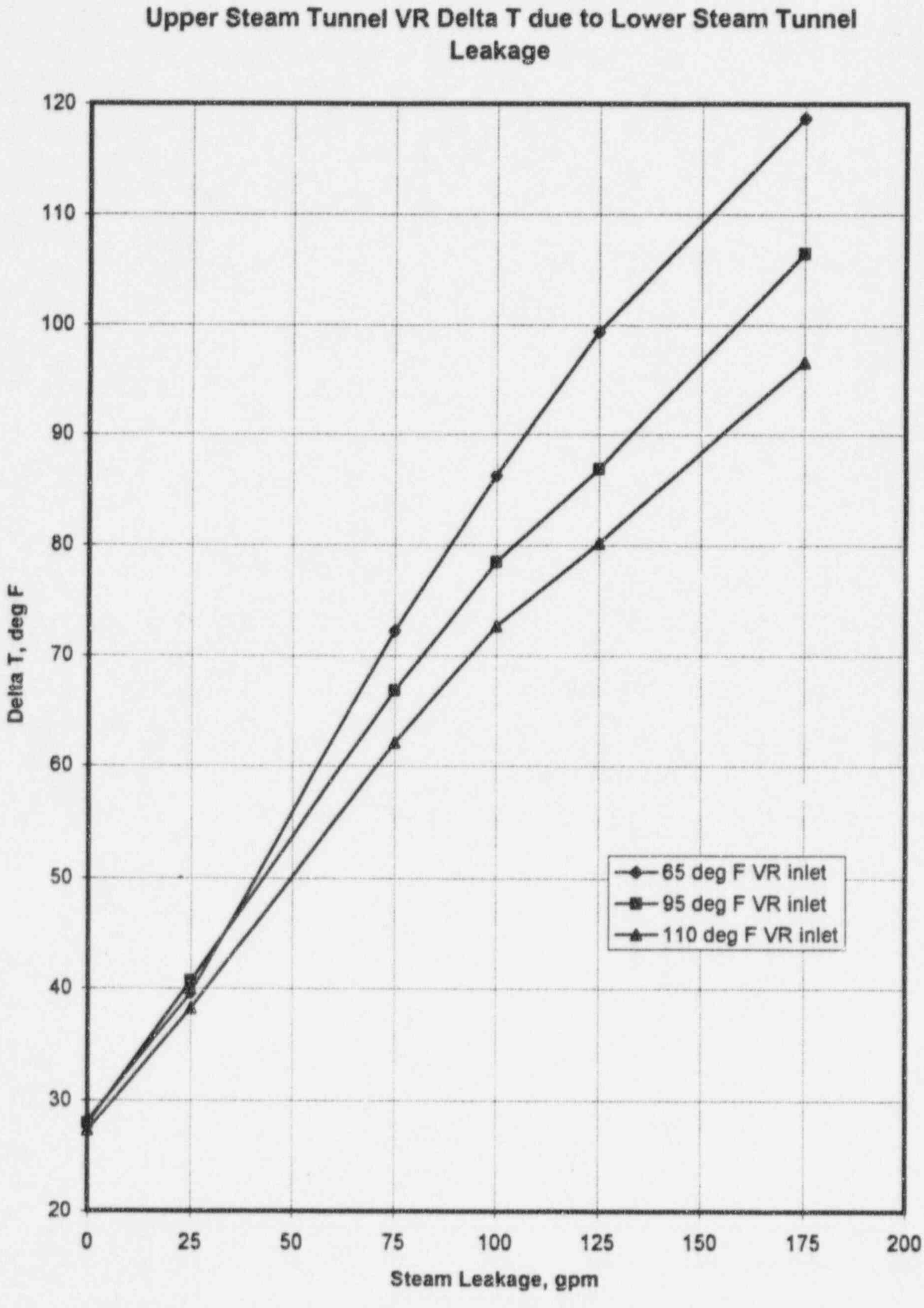

Upper Steam Tunnel Temperature due to Lower Steam Tunnel Leakage
210
200
190
180
170
160
150
140
130
120
110
100
90
80
Temperature, deg F
0
25
50
75
100
125
150
175
200
Steam Leakage, gpm
65 deg F VR inlet
95 deg F VR inlet
110 deg F VR inlet

Figure 3



Upper Steam Tunnel VR Delta T due to Lower Steam Tunnel Leakage
Delta T, deg F
120
110
100
90
80
70
60
50
40
30
20
0
25
50
75
100
125
150
175
200
Steam Leakage, gpm
65 deg F VR inlet
95 deg F VR inlet
110 deg F VR inlet

ATTACHMENT H

BASIS FOR DELETION OF MAIN STEAM LINE TUNNEL TEMPERATURE HIGH PRIMARY CONTAINMENT ISOLATION TRIP FUNCTION

The Bases for deletion of the Main Steam Line Tunnel Temperature - High Primary Containment Isolation instrumentation trip function from Technical Specification 3/4.3.2 is as follows:

The temperature and differential temperature instruments are not assumed to function to mitigate any accident described in Chapters 6 or 15 of the Updated Final Safety Analysis Report. The 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 steam line tunnel differential temperature and various other plant specific methods. Several of the BWR-6s have performed studies and analyses which supports the relocation of both the ambient and differential temperature instruments from the Technical Specifications. In addition, LaSalle County Station submitted an Amendment Request in July 1987 to delete these instruments from Technical Specifications. This Amendment Request was rejected by the staff based on the emphasis being primarily based on PRA study, however conceptually not unacceptable. In May 1989, a package was prepared to resubmit deletion of the high temperature and high differential temperature trips. However, due to time limits the package was revised to request longer allowed outage time for surveillances that affect main steam tunnel temperature and differential temperature instead of deletion. This amendment was approved for Units 1 and 2 on April 15, 1991 as amendments 77 and 61 to Unit 1 and 2 licenses NPF-11 and NPF-18, respectively.

The Ambient and Differential Temperature Isolation 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 Ambient and 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 Ambient and Differential Temperature Isolation Instruments are not used as part of a 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 ambient or differential temperature instruments. In addition, adequate redundancy is available to perform their functions by other methods.

A PRA study considered two factors in determining core damage frequency (CDF) impact: 1. The increase in CDF due to eliminating the automatic MSIV closure; and 2. the reduction in CDF by eliminating spurious MSIV closures.

1. Based on the low frequency for leaks to occur and the extremely low probability that operator action would not terminate the event in the available time frame, it was concluded that the increase in CDF due to elimination of the automatic trip function is negligible.

2. The results of the analysis indicate that the CDF would be reduced by an insignificant amount by elimination of all MSIV closure events. This analysis bounds the elimination of spurious events causing MSIV closure.

However, the differential temperature instruments will be retained for the purpose of an additional line break detection method that will provide earlier isolation of the steam lines and steam line drains than the other line break instrumentation. Together with Main Steam Line High flow and vessel low level, the differential temperature instruments adequate diversity is maintained, because previous evaluation of diversity of isolation parameters considered the ambient temperature and differential temperature isolations as one parameter in Table 5.2-8 of the LaSalle UFSAR.