TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	$\le$ 120 divisions of full scale	$\le$ 122 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-High, Setdown	$\le$ 15% of RATED THERMAL POWER	$\le$ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power - Upscale		
1) Two Recirculation Loop Operation		
a) Flow Biased	$\le$ 0.66W + 51% with a maximum of	$\le$ 0.66W + 54% with a maximum of
b) High Flow Clamped	$\le$ 113.5% of RATED THERMAL POWER	$\le$ 115.5% of RATED THERMAL POWER
2) Single Recirculation Loop Operation		
a) Flow Biased	$\le$ 0.66W + 45.7% with a maximum of	$\le$ 0.66W + 48.7% with a maximum of
b) High Flow Clamped	$\le$ 113.5% of RATED THERMAL POWER	$\le$ 115.5% of RATED THERMAL POWER
c. Fixed Neutron Flux-High	$\le$ 118% of RATED THERMAL POWER	$\le$ 120% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High	$\le$ 1043 psig	$\le$ 1063 psig
4. Reactor Vessel Water Level - Low, Level 3	$\ge$ 12.5 inches above instrument zero*	$\ge$ 11 inches above instrument zero*
5. Main Steam Line Isolation Valve - Closure	$\le$ 8% closed	$\le$ 12% closed
6. Main Steam Line Radiation - High	$\le$ 3 x full power background	$\le$ 3.6 x full power background
7. Primary Containment Pressure - High	$\le$ 1.69 psig	$\le$ 1.89 psig
8. Scram Discharge Volume Water Level - High	$\le$ 767' 5¼"	$\le$ 767' 5¼"
9. Turbine Stop Valve - Closure	$\le$ 5% closed	$\le$ 7% closed
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	$\ge$ 500 psig	$\ge$ 414 psig
11. Reactor Mode Switch Shutdown Position	N.A.	N.A.
12. Manual Scram	N.A.	N.A.

insert Attached page

*See Bases Figure B 3/4 3-1.

Attached page

INSERT FOR PAGE 2-4

13. Control Rod Drive

a.	Charging Water Header Pressure-Low	≥1157 psig	≥1134 psig
b.	Delay Timer	≤ 10 seconds	≤ 10 seconds

proposed change

~~DOCUMENT 0630r~~

New page

LIMITING SAFETY SYSTEM SETTING

BASES

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (CONTINUED)

13. Control Rod Drive (CRD) Charging Water Header Pressure - Low

The Hydraulic Control Unit (HCU) scram accumulator is precharged with high pressure nitrogen (N_2). When the Control Rod Drive (CRD) pump is activated, the pressurized charging water forces the accumulator piston down to mechanical stops. The piston is maintained seated against this mechanical stop with normal charging water pressure. If the charging water header pressure decreases below the N_2 pressure, such as would be the case with high leakage through the check valves of the CRD charging water lines, the accumulator piston would eventually rise off its stops. This results in a reduction of the accumulator energy and thereby degrades normal scram performance of the CRD's in the absence of sufficient reactor pressure.

The CRD low charging water header pressure trip setpoint initiates a scram at the charging water header pressure which assures the seating of the accumulator piston. With this trip setpoint, full accumulator capability, and therefore, normal scram performance, is assured at all reactor pressures. An adjustable time-delay relay is provided for each pressure transmitter/trip channel to protect against inadvertant scram due to pressure fluctuations in the charging line.

Four channels of pressure transmitter/trip unit combinations measure the charging water header pressure using one-out-of-two-twice logic. The trip function is active in STARTUP and REFUEL modes because reactor pressure may be insufficient to assist the CRD scram action.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each control rod scram accumulator shall be determined OPERABLE:

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

b. At least once per 18 months by:

1. Performance of a:

a) CHANNEL FUNCTIONAL TEST of the leak detectors, and

b) CHANNEL CALIBRATION of the pressure detectors, with the alarm setpoint 940 + 30, -0 psig on decreasing pressure.

2. Measuring and recording the time that each individual accumulator check valve maintains the associated accumulator pressure above the alarm setpoint with no control rod drive pump operating.

NO CHANGE

FOR REFERENCE ONLY

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place that trip system in the tripped condition* within 1 hour. The provisions of Specification 3.0.4 are not applicable.

b. With the the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within 1 hour and take the ACTION required by Table 3.3.1-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip functional unit shown in Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip system.

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*With a design providing only one channel per trip system, an inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.1-1 for that Trip Function shall be taken.

**If more channels are inoperable in one trip system than in the other, select that trip system to place in the tripped condition, except when this would cause the Trip Function to occur.

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
1. Intermediate Range Monitors:			
a. Neutron Flux - High	2	3	1
	3, 4	2	2
	$5^{(b)}$	3	3
b. Inoperative	2	3	1
	3, 4	2	2
	5	3	3
2. Average Power Range Monitor:$^{(c)}$			
a. Neutron Flux - High, Setdown	2	2	1
	3	2	2
	$5^{(b)}$	2	3
b. Flow Biased Simulated Thermal Power-Upscale	1	2	4
c. Fixed Neutron Flux-High	1	2	4
d. Inoperative	1, 2	2	1
	3	2	2
	5	2	3
3. Reactor Vessel Steam Dome Pressure - High	1, $2^{(d)}$	2	1
4. Reactor Vessel Water Level - Low, Level 3	1, 2	2	1
5. Main Steam Line Isolation Valve - Closure	$1^{(e)}$	4	4
6. Main Steam Line Radiation - High	1, $2^{(d)}$	2	5

NO CHANGE
FOR REFERENCE ONLY

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
7. Primary Containment Pressure - High	1, 2[(f)]	2[(g)]	1
8. Scram Discharge Volume Water Level - High	1, 2, 5[(h)]	2 2	1 3
9. Turbine Stop Valve - Closure	1[(i)]	4[(j)]	6
10. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	1[(i)]	2[(j)]	6
11. Reactor Mode Switch Shutdown Position	1, 2 3, 4 5	1 1 1	1 7 3
12. Manual Scram	1, 2 3, 4 5	1 1 1	1 8 9
13. Control Rod Drive			
a. Charging Water Header Pressure-Low	2 5[(h)]	2 2	1 3
b. Delay Timer	2 5[(h)]	2 2	1 3

NO CHANGES

FOR REFERENCE ONLY

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION STATEMENTS

ACTION 1 - Be in at least HOT SHUTDOWN within 12 hours.

ACTION 2 - Verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within 1 hour.

ACTION 3 - Suspend all operations involving CORE ALTERATIONS* and insert all insertable control rods within one hour.

ACTION 4 - Be in at least STARTUP within 6 hours.

ACTION 5 - Be in STARTUP with the main steam line isolation valves closed within 6 hours or in at least HOT SHUTDOWN within 12 hours.

ACTION 6 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to $\leq$ 140 psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER, within 2 hours.

ACTION 7 - Verify all insertable control rods to be inserted within 1 hour.

ACTION 8 - Lock the reactor mode switch in the Shutdown position within 1 hour.

ACTION 9 - Suspend all operations involving CORE ALTERATIONS,* and insert all insertable control rods and lock the reactor mode switch in the SHUTDOWN position within 1 hour.

*Except movement of IRM, SRM, or special movable detectors, or replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2.

No Change
For Reference Only

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* and during shutdown margin demonstrations performed per Specification 3.10.3.

(c) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.

(d) This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.10.1.

(e) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(f) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(g) Also actuates the standby gas treatment system.

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

(i) This function shall be automatically bypassed when turbine first stage pressure is $\leq$ 140 psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER.

(j) Also actuates the EOC-RPT system.

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*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High*	NA
b. Inoperative	NA
2. Average Power Range Monitor*	
a. Neutron Flux - High, Setdown	NA
b. Flow Biased Simulated Thermal Power-Upscale	$\leq$ 0.09**
c. Fixed Neutron Flux - High	$\leq$ 0.09
d. Inoperative	NA
3. Reactor Vessel Steam Dome Pressure - High	$\leq$ 0.55
4. Reactor Vessel Water Level - Low, Level 3	$\leq$ 1.05
5. Main Steam Line Isolation Valve - Closure	$\leq$ 0.06
6. Main Steam Line Radiation - High	NA
7. Primary Containment Pressure - High	NA
8. Scram Discharge Volume Water Level - High	NA
9. Turbine Stop Valve - Closure	$\leq$ 0.06
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	$\leq$ 0.08#
11. Reactor Mode Switch Shutdown Position	NA
12. Manual Scram	NA
13. Control Rod Drive	
a. Charging Water Header Pressure - Low	NA
b. Delay Timer	NA

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel.

**Not including simulated thermal power time constant.

#Measured from start of turbine control valve fast closure.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors				
a. Neutron Flux - High	S/U[b], S	S/U[c], W	R	2
	S	W	R	3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor:[f]				
a. Neutron Flux - High, Setdown	S/U[b], S	S/U[c], W	SA	1, 2
	S	W	SA	3, 5
b. Flow Biased Simulated Thermal Power-Upscale	S, D[g]	S/U[c], W	W[d][e], SA, R[h]	1
c. Fixed Neutron Flux - High	S	S/U[c], W	W[d], SA	1
d. Inoperative	NA	W	NA	1, 2, 3, 5
3. Reactor Vessel Steam Dome Pressure - High	NA	M	Q	1, 2
4. Reactor Vessel Water Level - Low, Level 3	S	M	R	1, 2
5. Main Steam Line Isolation Valve - Closure	NA	M	R	1
6. Main Steam Line Radiation - High	S	M	R	1, 2
7. Primary Containment Pressure - High	NA	M	Q	1, 2

NO CHANGES
FOR REFERENCE ONLY

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High	NA	M	R	1, 2, 5
9. Turbine Stop Valve - Closure	NA	M	R	1
10. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	NA	M	R	1
11. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	M	NA	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) The IRM, and SRM channels shall be determined to overlap for at least 1/2 decades during each startup and the IRM and APRM channels shall be determined to overlap for at least 1/2 decades during each controlled shutdown, if not performed within the previous 7 days.
(c) Within 24 hours prior to startup, if not performed within the previous 7 days.
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2%. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.
(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.
(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.
(g) Measure and compare core flow to rated core flow.
(h) This calibration shall consist of verifying the 6 ± 1 second simulated thermal power time constant.

13. Control Rod Drive				
a. Charging Water Header Pressure	NA	M	R	2,5
b. Delay Timer	NA	M	R	2,5

CONTROL RODS (Continued)

insert following page

Control rod coupling integrity is required to ensure compliance with the analysis of the rod drop accident in the FSAR. The overtravel position feature provides the only positive means of determining that a rod is properly coupled and therefore this check must be performed prior to achieving criticality after completing CORE ALTERATIONS that could have affected the control rod drive coupling integrity. The subsequent check is performed as a backup to the initial demonstration.

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE.

The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the event of a housing failure. The amount of rod reactivity which could be added by this small amount of rod withdrawal is less than a normal withdrawal increment and will not contribute to any damage to the primary coolant system. The support is not required when there is no pressure to act as a driving force to rapidly eject a drive housing.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than 20% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RSCS and RWM to be OPERABLE when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER provides adequate control.

The RSCS and RWM provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section 15.4.9 of the FSAR and the techniques of the analysis are presented in a topical report, Reference 1, and two supplements, References 2 and 3.

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods.

Insert to page B 3/4 1-3

In addition, the automatic CRD charging water header low pressure scram (see Table 2.2.1-1) initiates well before any accumulator loses its full capability to insert the control rod. With this added automatic scram feature, the surveillance of each individual accumulator check valve is no longer necessary to demonstrate adequate stored energy is available for normal scram action.

Proposed change

ATTACHMENT 3

COMMONWEALTH EDISON COMPANY

LASALLE COUNTY STATION UNIT 2

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 10CFR50.92, operation of LaSalle County Station Unit 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 this change to the Technical Specifications provides greater assurance that the scram function will mitigate the consequences of a postulated accident. This CRD charging water header low pressure scram is discussed in the FSAR and in LaSalle County Station Safety Evaluation Report supplement 7. The revised setpoints are based solely on reduced uncertainties allowed by reducing the calibrated range of the pressure transmitter and trip units.

2) Create the possibility of a new or different kind of accident from any previously evaluated because this change does not eliminate any previously required scram function but adds an additional one to greater ensure automatic control rod insertion capability under all plant operating conditions.

3) Involve a significant reduction in the margin of safety because this change maintains or increases the likelihood that proper control rod scram capability will be available during all plant conditions.

Based on the preceding discussion, it is concluded that the proposed system change clearly falls within all acceptable criteria with respect to the system or component, the consequences of previously evaluated accidents will not be increased and the margin of safety will not be decreased. Therefore, based on the guidance provided in the Federal Register and the criteria established in 10CFR50.92(e), the proposed change does not constitute a significant hazards consideration.