Commonwealth Edison
One First National Plaza, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690

February 10, 1981

Mr. B.J. Youngblood, Chief
Licensing Branch 1
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: LaSalle County Station Units 1 & 2, Instrumentation and Control Systems Resolution of SER Open Issues, NRC Docket Nos. 50-373/374
LOD 81-40-10

Dear Mr. Youngblood:

Attached for your review are supplemental materials submitted in response to NRC Staff questions related to the following Instrumentation and Control Systems issues:

1. Test Techniques - see Enclosure 1
2. Setpoints - see Enclosure 2
3. Separation Criteria - see Enclosure 3
4. Testing of Low-Low Setpoint Logic - see Enclosure 4
5. RG 1.97 and Safe Shutdown Indication - see Enclosure 5
6. RBM System - see Enclosure 6
7. Use of Non-Safety Grade Equipment

Each of these issues is discussed in the enclosure to this letter as referenced above. It is judged that these supplemental materials provide the technical basis for resolving these issues. This judgement is based on our review of the subject areas and detailed discussions conducted with the NRC Staff during the week of February 2, 1981. Item 7, although discussed at sor ngth, is not addressed in the enclosures inasmuch as the Staff has nc decided whether additional information in addition to technical spec ions already committed is required.

In the event yc have any further questions on these matters, please direct them to this office.

Very truly yours,

L.O. DelGeorge

L.O. DelGeorge
Nuclear Licnesing Administrator

Enclosures 1-6

cc: NRC Resident Inspector-LSCS

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ENCLOSURE 1

TEST TECHNIQUES

In order to address concerns raised by the NRC Staff relative to the technique used in the performance of periodic instrumentation surveillances on LaSalle County Unit 1, a detailed deliniation of all applicable test procedures was provided in response to Q 031.284. In addition, the testing frequencies associated with these tests as defined in the LaSalle County Technical Specifications was reviewed with the Staff. At the conclusion of that review, the Staff requested a clarification of the input to Q 031.284 Section E, "Logic System Functional Tests Covered By Technical Specifications".

This question was reviewed in greater depth by Commonwealth Edison and the following information provided to the Staff:

1. Logic System Functional Tests, as described in Section 4.3- of the Technical Specifications are performed on an 18 month interval during refueling outage.

2. Item Q 031.284 E.1, "RPS Instrumentation" does not require circuit disturbance of the types discussed in Q 031.284 to accomplish the required periodic surveillance.

3. Item Q 031.284 E.2, "Isolation Instrumentation" was reassessed and it was determined that fuse pulling was required only for Item 1.b (Hi Drywell Pressure) in Table 4.3.2-1 and that an annunciator alarm which is received on removal of the fuse which must be cleared as part of the return of the system to normal.

4. Item Q 031.284 E.3, "ECCS Actuation Instrumentation" and E.4, "RCIC Actuation Instrumentation" were reassessed and it was determined that fuse pulling was required for Items a.1.f, a.1.i, a.2.c, a.2.g, b.1.d and b.1.g. These tests require check of time delay relays and manual push buttons, An annunciator alarm is received on removal of the fuse which must be cleared as part of the return of the system to normal.

This information was discussed with Mr. D. Thatcher of ICSB on February 6, 1981 and is reported here for documentation. Question Q 031.284 will be modified to document this information in a future amendment to the FSAR

ENCLOSURE 2

SETPOINTS

At the review meeting between Commonwealth Edison and the NRC staff on December 17, 1981, the applicant was requested to provide an amendment to Tables 7.1-2, 7.3-1, 7.3-2 and 7.6-1 of the FSAR which would identify the device range for instruments delineated therein. These tables have been reviewed, and changes made as requested. A copy of the prosed amendment to those tables is provided as an attachment to this enclosure. Formal documentation of these amended tables will be made in a future amendment to the FSAR.

LA SALLE 13 Feb 1981

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TABLE 7.3-1

ECCS ACTUATION INSTRUMENTATION SETPOINTS (1)

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE	ANALYTIC OR DESIGN BASIS LIMIT	ACCURACY	CALIBRATION	DESIGN-BASIS ALLOWABLE	Device Range
	HPCS							
(1)	Reactor Water Level - Low, Level #2	-50 in.	-57 in.	-70 in.	6.0 in.	2.0 in.	7.0 in.	-150/0/+150 in
(2)	Drywell Pressure - High	1.69 psig	1.89 psig	2.0 psig	0.04 psi	0.04 psi	0.2 psi	0.2-6 psi
(3)	Condensate Storage Tank Level - Low	765' 7"	765 ft. 3"	DB	0.5 in.	0.5 in.	1.0 in.	-1.0/0/+1.0 in
(4)	Suppression Pool Water Level - High	700 ft. 1 in.	700 ft. 2 in.	DB	0.5 in.	0.5 in.	1.0 in.	-1.0/0/+1.0 in
	ADS							
(5)	Drywell Pressure - High	1.69 psig	1.89 psig	2.0 psig	0.04 psi	0.04 psi	0.2 psi	0.2-6 psi
(6)	Reactor Water Level - Low, Level #1	-129 in.	-136 in.	-149 in.	6.0 in.	2.0 in.	7.0 in.	-150/0/+60 in
(7)	ADS Timer	105 seconds	117 seconds	120 seconds	2.0 sec.	2.0 sec.	12.0 sec	4-120 sec
(8)	LPCS Pump Discharge Pressure - Permissive	146 psig	136 psig	125 psig	5.0 psi	2.0 psi	10.0 psi	10-340 psig

See Figure ~~5.2-5.~~ ~~Table VI for elevation correlation chart.~~

(1) See Technical Specifications for operational instrument specifics

TABLE 7.3-1 (Cont'd)

add

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE	ANALYTIC OR DESIGN BASIS LIMIT	ACCURACY	CALIBRATION	DESIGN-BASIS ALLOWABLE	DEVICE RANGE
(9)	RHR (LPCI Mode) Pump Discharge Pressure - Permissive	~~125~~ 119 psig	106 psig	100 psig	2.0 psi	2.0 psi	8.0 psi	10-240 psig
	LPCS							
(10)	Reactor Vessel Water Level - Low, Level #1	-129 in.	-136 in.	-~~149~~ 150 in.	6.0 in.	2.0 in.	7.0 in.	-150/0/+60 in
(11)	Injection Valve ΔP - High	729 psid	709 psid	700 psid	3.0 psi	3.0 psi	20.0 psi	0-800 psi
(12)	Drywell Pressure - High	1.69 psig	1.89 psig	2.0 psig	0.04 psi	0.04 psi	0.2 psi	0.2-6 psi
	RHR (LPCI Mode)							
(13)	Drywell Pressure - High	1.69 psig	1.89 psig	2.0 psig	0.04 psi	0.04 psi	0.2 psi	0.2-6 psi
(14)	Reactor Vessel Water Level - Low, Level #1	-129 in.	-136 in.	-149 in.	6.0 in.	2.0 in.	7.0 in.	-150/0/+60 in
(15)	Injection Valve ΔP - High	729 psid	709 psid	700 psid	3.0 psi	3.0 psi	20.0 psi	0-800 psi

TABLE 7.3-2

PRIMARY CONTAINMENT AND REACTOR VESSEL

ISOLATION ACTUATION INSTRUMENT SETPOINTS

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	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALVE	ANALYTIC OR DESIGN-BASIS LIMIT	ACCURACY	CALIBRATION	DESIGN-BASIS ALLOWANCE	DEVICE RANGE
	Reactor Core Isolation Cooling							
(1)	RCIC Steamline Flow - High	290% (~~175~~ 178 in.)	~~297%~~ 295 (~~187~~ 185 in.)	300% (191 in.)	1 in.	1 in.	6 in.	-300/0/+300 in
(2)	RCIC Steam Supply Pressure - Low	57 psig	53 psig	DB	1.0 psi	1.0 psi	2.0 psi	4-100 psig
(3)	RCIC Pipe Routing Area Temperature - High	* 206°F	* 212°F	*	2.0° F	1.0° F	6.0° F	50-350°F
(4)	RCIC Pipe Routing Area Δ Temperature - High	* 130°F	* 136°F	*	3.0° F	0.5° F	3.0° F	0-150°F
(5)	RCIC Turbine Exhaust Diaphragm Pressure - High	10.0 psig	20.0 psig	DB	1.0 psi	1.0 psi	4.0 psi	0.5-80 psig
(6)	RCIC Equipment Room Temperature - High	* 160°F-200°F	* 166°F-206°F	*	2.0° F	1.0° F	6.0° F	50-350°F
~~(7)~~	~~RCIC Equipment Room Δ Temperature - High~~	~~*~~	~~*~~	~~*~~	~~3.0° F~~	~~0.5° F~~	~~3.0° F~~	
	Shutdown Cooling Isolation							
(8)	Reactor Vessel Water Level - Low, Level #3	12.5 in.	11.0 in.	7.5 in.	1.5 in.	0.5 in.	1.5 in.	0-60 in

* During preoperational testing, the trip setpoints will be set at 40° F above space ambient temperature. Final setpoints will be established based upon operational data after calibration of detectors and modules.

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TABLE 7.3-2 (Cont'd)

	FUNCTIONAL UNIT		TRIP SETPOINT	ALLOWABLE VALVE	ANALYTIC OR DESIGN-BASIS LIMIT	ACCURACY	CALIBRATION	DESIGN-BASIS ALLOWANCE	DEVICE RANGE
(9)	Reactor ~~Steam Dome~~ Pressure - High	Loop A	135 psig	145 psig	161 psig	5.0 psi	2.0 psi	10 psi	10-240 psig
		Loop B	135 psig	145 psig	161 psig	5.0 psi	2.0 psi	10 psi	0-500 psig
	Reactor Water Cleanup System								
(10)	Δ Flow - High		70 gpm	87.5 gpm	104.5 gpm	11.5 gpm	13 gpm	17.5 gpm	0-100 gpm
(11)	Area Temperature - High		*185°F	*191°F	*	2.0° F	1.0° F	6.0° F	50-350 °F
(12)	Area Ventilation Temp. ΔT-High		* 95°F	* 101°F	*	3.0° F	0.5° F	3.0° F	0-150 °F
(13)	Reactor Vessel Water Level - Low, Level #2		-50 in.	-57 in.	-70 in.	6.0 in.	2.0 in.	7.0 in.	-150/0/+60 in
	Residual Heat Removal								
(14)	RHR Flow - High		123 in.	128 in.	DB	6 in.	3 in.	5 in.	0-350 in
(15)	RHR Equipment Area Temperature - High		*160°F	*166°F	*	2° F	1° F	6° F	50-350°F
(16)	RHR Equipment Area Δ Temperature - High		* 50°F	* 56°F	*	3° F	0.5° F	3° F	0-150°F
	Primary Containment								
(17)	Reactor Vessel Water Level - Low, Level #3		12.5 in.	11.0 in.	7.5 in.	1.5 in.	0.5 in.	1.5 in.	0-60 in
(18)	Reactor Vessel Water Level - Low, Level #2		-50 in.	-57 in.	-70 in.	6.0 in.	2.0 in.	7.0 in.	-150/0/+60 in

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TABLE 7.3-2 (Cont'd)

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALVE	ANALYTIC OR DESIGN-BASIS LIMIT	ACCURACY	CALIBRATION	DESIGN-BASIS ALLOWANCE	DEVICE RANGE
(19)	Drywell Pressure - High	1.69 psig	1.89 psig	2.0 psig	0.04 psi	0.04 psi	0.2 psi	0.2-6 psi
(20)	Main Steamline Radiation - High	3.0 x full power background	3.6 x full power background	DB	20%	2%	10%	1-10^6 mR/hr
(21)	Main Steamline Pressure - Low	854 psig	834 psig	825 psig	3.0 psi	3.0 psi	20.0 psi	200-1200 psi
(22)	Main Steamline Flow - High	111 psid	116 psid	123 psid	0.5 psid	0.5 psid	3.0 psi	15/0/+150 psid
(23)	Main Steamline Tunnel Temperature - High	* 140°F	* 146°F	*	2° F	1° F	6° F	50-350°F
(24)	Main Steamline Δ Temperature - High	* 20°F	* 26°F	*	3° F	0.5° F	3° F	0-150°F
(25)	Condenser Vacuum - Low	* 24.5 in Hg	* 23 in Hg	*	0.3 in. Hg	0.3 in. Hg	0.6 in. Hg	0.8-29.2 in Hg
	Secondary Containment							
(26)	Reactor Building Exhaust Rad - High	* 10 mR/hr ~~130% A.L.~~	15 mR/hr ~~150% A.L.~~	*	10%	2%	12%	0.01-100 mR/hr
(27)	Drywell Pressure - High	1.69 psig	1.89 psig	2.0 psig	0.04 psi	0.04 psi	0.2 psi	0.2-6 psi
(28)	Reactor Vessel Water Level - Low, Level #3	12.5 in.	11.0 in.	7.5 in.	1.5 in.	0.5 in.	1.5 in.	0-60 in

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TABLE 7.3-4

REACTOR PROTECTION SYSTEM INSTRUMENT SETPOINTS

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE	ANALYTIC OR DESIGN-BASIS LIMIT	ACCURACY	CALIBRATION	DESIGN-BASIS ALLOWANCE	Device RANGE
(1)	Intermediate Range Monitor Neutron Flux - Upscale	120 Divisions	122 Divisions	DB	0.1%	0.5%	2.0%	2% to Full Scale.
(2)	Average Power Range Monitor Neutron Flux - Upscale (Not Run Mode)	15%	20%	25%	0.1%	0.5%	2.0%	N/a
(3)	Average Power Range Monitor, Simulated Thermal Power - Upscale	(.66w + 51%) (113.5% max.)	(.66w + 54%) (115.5% max.)	117%	0.1%	0.5%	2.0%	N/a
(4)	Average Power Range Monitor, - Upscale (Run Mode)	118%	120%	121%	0.1%	0.5%	2.0%	N/a
(5)	Reactor Vessel ~~Steam Dome~~ Pressure - High	1043 psig	1063 psig	1071 psig	3.0 psi	3.0 psi	20.0 psi	200-1200 psi
(6)	Reactor Vessel Water Level - Low, Level #3*	12.5 in.	11.0 in.	7.5 in.	1.5 in.	0.5 in.	1.5 in.	0-60 IN
(7)	Main Steamline Isolation Valve - Closure	94% open	93% open	90% open	1.0%	1.0%	1.0%	—
(8)	Main Steamline Radiation - High	3.0 x full power background	3.6 x full power background	DB	20%	2%	10%	1×10^6 mR/hr

*Refers to instrument zero wheihc is 527.5 inches above vessel zero.

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TABLE 7.3-4 (Cont'd)

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE	ANALYTIC OR DESIGN-BASIS LIMIT	ACCURACY	CALIBRATION	DESIGN-BASIS ALLOWANCE	Add Device Range
(9)	Drywell Pressure - High	1.69 psig	1.89 psig	2.0 psig	0.04 psi	0.04 psi	0.2 psi	0.2-6 psi
(10)	Scram Discharge Volume Water Level - High	*	*	*	0.5 in.	0.5 in.	1.0 in.	
(11)	Turbine Stop Valve - Closure	95% open	93% open	90% open	1.0%	1.0%	2.0%	n/a
(12)	Turbine Control Valve - Fast Closure, Trip Oil Pressure - Low	500 psig	414 psig	400 psig	5.0 psi	5.0 psi	30.0 psi	n/a

Note: All reactor water levels are referenced to instrument zero at 527.5 inches above the vessel ~~zones~~ zero

* To be determined from preoperational testing.

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TABLE 7.3-5

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE	ANALYTIC OR DESIGN BASIS LIMIT	ACCURACY	CALIBRATION	DESIGN DRIFT ALLOWANCE	Device Range
	Average Power Range Monitor							
(1)	Neutron Flux - Upscale (flow referenced)	(0.66w + 42%)	(0.66w + 45%)	DB	1.0%	1.0%	3.0%	n/a
(2)	Neutron Flux - Downscale	5%	3%	DB	0.1%	0.5%	2.0%	"
(3)	Neutron Flux - Upscale (Not Run Mode)	12%	14%	DB	0.1%	0.5%	2.0%	"
	Rod Block Monitor							
(4)	Upscale	(0.66w + 40%)	(0.66w + 43%)	111.8%	1.0%	1.0%	3.0%	n/a
(5)	Downscale	5%	3%	DB	0.1%	0.5%	2.0%	"
	Source Range Monitors							
(6)	Upscale	2×10^5 cps	5×10^5	DB	0.1%*	0.5%*	2.0%*	$10^{-1} - 10^6$ cps
(7)	Downscale	3 cps	2 cps	DB	0.1%*	0.5%*	2.0%*	
	Intermediate Range Monitor							
(8)	Upscale	108/125	110/125	DB	2.0%	0.5%	2.0%	2% to Full Scale
(9)	Downscale	5/125	3/125	DB	2.0%	0.5%	2.0%	

* Equivalent Linear Full Scale.

POOR ORIGINAL

ENCLOSURE 3

SEPARATION CRITERIA

The criteria used in the separation and physical independence design for LaSalle County Station Units 1 & 2 is described in Section 8.3.1 of the FSAR. In response to inquiries from the NRC Staff a reassessment of these criteria has been performed with the intention of establishing the degree of conformance to Regulatory Guide 1.75 "Physical Independence of Electric Systems" which was published (Feb. 3, 1974) after the LaSalle County Construction Permit was issued (Sept. 1973) and revised (Jan., 1975) after the LaSalle County final design basis was implemented.

The discussion of RG 1.75 contained in Appendix B of the FSAR is being revised to clarify the bases upon which it is judged that the LaSalle County design provides a technically acceptable alternative to the regulatory position of RG 1.75. This revision was discussed with the NRC Staff on February 6, 1981 and judged to be tentatively acceptable pending adequate documentation. A copy of the proposed revision to Appendix B is attached and will be formally documented in a future amendment to the FSAR.

REGULATORY GUIDE 1.75

Initial Issue: Revision 0, February 1974

Current Issue: Revision 1, January 1975

La Salle C.P. Issued: September 10, 1973

PHYSICAL INDEPENDENCE OF ELECTRIC SYSTEMS

Regulatory Guide 1.75 describes a method acceptable to the NRC for assuring physical independence of the circuits and electric equipment comprising or associated with Class 1E power systems, protection systems, systems actuated or controlled by the protection system, and auxiliary or supporting systems that must be operable for the protection systems to perform their safety-related functions.

Regulatory Guide 1.75 was issued after the La Salle County Station construction permit. However, the LSCS design meets MOST ~~the intent~~ of the regulatory positions, as described in Subsection 8.3.1. Aside from the two ~~three~~ exceptions noted below (Positions 1, ~~2~~ and 10) the LSCS design complies with this guide. POSITION 2 IS NOT APPLICABLE.

The LSCS design differs with Position 1 in that the basis for precluding the use of a fault-current-actuated interrupting device as an isolation device within the context of Regulatory Guide 1.75 is questionable. This conclusion seems to preclude the use of a circuit breaker as an isolating device and thus prevents achieving a practical design of an electrical auxiliary system. A qualified (Class 1E) circuit breaker could otherwise fulfill the literal interpretation of Section 3.8 by preventing (by its successful interruption of fault current) a malfunction in one section of a circuit from causing unacceptable influences (i.e., failures) in other sections of the circuit (or other circuits). To preclude Class 1E circuit breakers from these applications casts doubt upon the safety-related portions of the auxiliary power systems in all nuclear plants. Such systems invariably utilize Class 1E metal-clad switchgear and molded case circuit breakers as circuit protection devices.

SEE INSERT ~~The LSCS design fully complies with Positions 2, 3, 4, 5, and 6. Position 7 refers to Section 4.6.2 of IEEE 384-1974 concerning non-class 1E instrumentation and control circuits not being required to be separated from associated circuits. The LSCS design is in agreement with Section 4.6.2 of IEEE 384-1974 in that non-class 1E instrumentation and control circuits are not required to be separated from associated~~

INSERT

The La Salle County Station design complies with Positions 3 thru 9. With respect to the separation of associated circuits, the La Salle County Station design complies with Section 4.5 of IEEE-384-1974. Although separation of associated circuits is, in some cases, slightly different that that dictated in Sections 5.13 and 5.62 of IEEE-384, Class 1E circuits are not degraded below an acceptable level for the following reasons:

a. Cables associated with one safety-related division are never routed in cable trays or conduit containing cables of a redundant safety-related division. This is true for all general plant areas. Lesser separation than that dictated by IEEE-384 occurs only within control panels located in the control room and auxiliary equipment room.
b. All cables used to interconnect associated circuits are the same high quality as that utilized in Class 1E circuits, i.e., all associated cables comply with the requirements of IEEE 383-1974. Therefore, this cable has been proven to be highly fire retardant by testing.
c. A lesser separation than that dictated by Sections 5.13 and 5.62 of IEEE-384 is limited to control and instrumentation circuits which, by their very nature, are low energy circuits. Control circuits are generally 120V a.c. or 125V d.c., whereas, the insulation rating of the cable utilized at La Salle County is 600V.
d. There are not power cables in contact with the control and instrumentation cables in the cable spreading area or in the control room and auxiliary equipment room. Also, there are no high energy sources located within control panels installed in these areas.
e. Fire stops are installed in the bottom entrances of all control panels.

With respect to the separation of non-Class 1E from Class 1E control and instrumentation circuits, La Salle County Station complies with Section 4.6.2 of IEEE-384. Although the separation of non-Class 1E from Class 1E control and instrumentation circuits is, in some cases, less than that dictated by Sections 5.1.3 and 5.6.2 of IEEE-384, these circuits have been analyzed to show that Class 1E circuits are not degraded below an acceptable level for the following reasons:

a. Non-Class 1E cables are routed in separate cable trays from Class 1E and associated cables in general plant areas.
b. Non-Class 1E cables which come in close proximity to Class 1E and associated cables at one end do not come in contact with redundant Class 1E or associated circuits at their other end. (This has been confirmed by a study of the La Salle County Station installation.)
c. All cables used to interconnect associated circuits are the same high quality as that utilized in non-Class 1E circuits, i.e., all associated cables comply with the requirements of IEEE 383-1974. Therefore, this cable has been proven to be highly fire retardant by testing.
d. A lesser separation than that dictated by Sections 5.1.3 and 5.6.2 of IEEE-384 is limited to control and instrumentation circuits which by their very nature are low energy circuits. Control circuits are generally 120V a.c. or 125V d.c., whereas, the insulation rating of the cable utilized at La Salle County is 600V.
e. There are no power cables in contact with the control and instrumentation cables in the cable spreading area or in the control room and auxiliary equipment room. Also, there are no high energy sources located within control panels installed in these areas.
f. Fire stops are installed in the bottom entrances of all control panels.

~~circuits since these are low energy circuits and all cabling used in these circuits are fully qualified to the requirements of IEEE 383-1974~~.

~~The LSCS design fully complies with Positions 8 and 9~~. Position 10 refers to Section 5.1.2 of IEEE 384-1974 concerning cable and raceway identification. The LSCS design utilizes cable trays with permanent colored identification markers at each routing point which are assigned an alphanumeric code per Table 8.3-6 of the LSCS-FSAR. Each cable is assigned a number and segregation code. This information is placed on a colored tag, of permanent design, which is affixed to each end of the cable. A similar tag is also affixed to the cable where it enters and exits a penetration.

The LSCS design complies with Positions 11, 12, 13, 14, 15, and 16.

ENCLOSURE 4

TESTING OF LOW-LOW SETPOINT SYSTEM LOGIC

The low-low setpoint system logic which is being installed on LaSalle County Units 1 & 2 as a product improvement is intended to reduce the number of relief valve repetative actuations thereby enhancing S/RV load margins. This system is being functionally tested as a part of the LaSalle County Initial Test program to assure proper system operation. At the request of the NRC Staff, a copy of the system test program was provided on February 6, 1981. It is our understanding that the NRC will review this information and determine whether periodic surveillance of this system, which is not required to satisfy the plant design basis, need be considered. In the event periodic surveillance is required, it is our understanding that it will be limited to a logic system functional test on an 18 month (refueling outage) cycle.

ENCLOSURE 5

POST-ACCIDENT MONITORING
AND
SAFE-SHUTDOWN INDICATION

The LaSalle County Station Unit 1 & 2 design has been upgraded to address the issue of adequate monitoring during and following an accident. Significant improvements have already been implemented to improve monitoring of Engineered Safety Features (FSAR Section 7.8), Post-Accident Tracking (FSAR Section 7.5.1.2) and other instrumentation systems required for safety which are related to the subject of monitoring during and following an accident (FSAR Section 7.6). In addition significant improvements have also been made in response to NUREG-0737 Item II.F.1 related to Post-Accident Monitoring (FSAR Appendix L.29). Activities are currently underway to assess the need to provide additional instrumentation to satisfy the requirements of NUREG-0737 Items II.D.2 and II.F.2(1) for which criterion applicable to BWR facilities has not been determined as yet by the NRC. In addition, related areas addressed in NUREG-0737 Items III.A.1.2 and III.A.2 on Emergency Preparedness are also being reviewed and will be resolved on a schedule consistent with that ultimately prescribed by the NRC, conditioned only on unforeseen difficulties in procuring and installing additions or modifications to in-place instrumentations.

In this regard, Commonwealth Edison will take guidance from the position defined in Regulatory Guide 1.97, "Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident". It is expected that the review of the adequacy of the LaSalle County instrumentation design will be conducted in the context of NUREG-0737 Items II.D.2, II.F.2 (1), III.A.1.2 and III.A.1 and will be resolved on the schedule established for conformance with that document or future revisions to it. In any event, resolution of these issues prior to LaSalle County Unit 1 Fuel loading is required only insofaras the quidance presently defined in NUREG-0737 require.

One additional area in which the LaSalle County design has been modified to provide additional information related to safe-shutdown indication is the Rod Position Indication System (RPIS). With the agreement of the NRC Staff a modification to this system will be made as soon as possible (current schedules indicate that this modification can be completed prior to Unit 1 fuel loading) to provide continuous unterruptible power to the RPIS. This modificaiton and proposed amendments to the FSAR are presented in greater detail in the attachment to this enclosure.

radiation monitor is displayed on an eight-decade meter and on separate stripchart recorders located in the control room.

Redundancy

The subsystem utilizes a redundant instrumentation channel so that a single failure cannot prevent subsystem operation.

Separation

Each of the redundant pairs of gamma-sensitive instrumentation is physically separated from the other and is powered from a separate power bus.

Inspection and Testing

A built-in source of current is provided with each radiation monitor for test purposes to provide a point reading equivalent to 10^5 R/hr. In addition, the operability of each monitoring channel can be routinely verified by comparing the outputs of the channels at any time.

Environmental Considerations

The gross gamma monitoring equipment read-outs in the control room. See Appendix M for a description of the reactor building and control room environment.

Operational Considerations

The gross gamma subsystem is operational at all times during normal and accident conditions except when taken out of service for calibration.

This subsystem covers the range of 10^0 to 10^8 R/hr, which is greater than the dose rates in the H_2 and O_2 sample lines following the loss-of-coolant accident.

7.5.1.3 Shutdown, Isolation, and Core Cooling Indication

The information furnished to the control room operator permits him to assess reactor shutdown, isolation, and availability of emergency core cooling following the postulated accident.

a. Operator verification that reactor shutdown has occurred may be made by observing one or more of the following indications:

SEE INSERT FOR 1. AND 2. ~~1. Control rod status lamps indicating each rod fully inserted. (See Figure 7.5-1.) The power source is one of the~~

INSERT 1

INSERT FOR SUBSECTION 7.5.1.3 a

1. Control rod status lamps indicating each rod fully inserted. See Figure 7.5-1. The power source for these lights is the non Safety-related Source shown on Figures 1E-1-4206AE and 1E-1-4206AF (in Section 1.7).

2. Computer printout of rod position information indicating actual rod position. The power source for both the computer and the rod position indication logic panel (1H13-P615) is the uninterruptible power supply. The power source to the RPI logic panel (1H13-P615) is shown on Figure 1E-1-4213AC (in Section 1.7).

~~instrument a-c buses.~~

3 ~~/~~. Control rod scram pilot valve status indicating open valves. See Figure 7.5-1. The power sources are RPS MG sets. See Figure 1E-1-4215AJ (Section 1.7).
NOTE: RPS MG SETS ARE POWERED FROM MOTOR CONTROL CENTERS WHICH ARE POWERED FROM THE EMERGENCY DIESEL GENERATORS ON LOSS OF OFFSITE POWER.

4 ~~/~~. Neutron monitoring power range channels and recorders downscale. See Figure 7.5-1. The power sources are RPS MG sets, WHICH ARE ULTIMATELY POWERED BY THE EMERGENCY DIESEL GENERATOR ON LOSS OF OFFSITE POWER.

5 ~~/~~. Annunciators for reactor protection system variables and trip logic in the tripped state. See Figure 7.5-1. The power source is dc from a station battery.

b. The operator may verify reactor isolation by observing one or more of the following indications:

1. Isolation valve position lamps indicating valve closure. See Figure 7.5-2. The power source is the same as for the associated motor operator.

2. Main steamline flow indication downscale. See Figure 7.5-1. The power source is instrument ac from one of the standby a-c buses.

c. Operation of the emergency core cooling and the RCIC system following an accident may be verified by observing the following indications:

1. Flow and pressure indications for each emergency core cooling system. See Figure 7.5-2. The power sources are independent and from the same standby buses as the driven equipment.

2. RCIC isolation valve position indicating open valves. See Figure 7.5-2. The power source is from the same bus as the valve motive power.

3. Injection valve position lights indicating either open or closed valves. See Figure 7.5-2. The power source is the same as the valve motor.

4. Relief valve position status by open or closed indicator lamps. See Figure 7.5-2. The power source is the same as for the pilot solenoid, 120 Vac from standby a-c systems.

detect pressurization upward on the drywell floor due to the suppression pool bypass phenomena, in order to insure that design limits of the drywell floor are not exceeded. These transmitters feed redundant divisional indicators in the control room to provide instantaneous pressure readings to the operator. In addition, for the non-accident case, an alarm is provided in the control room to call the operator's attention to these indicators if there is an abnormal pressure reading.

The transmitters are qualified to IEEE 323-1971 standards, and each loop in the redundant pair is powered from emergency a-c power, which is capable of being powered by the diesel generators.

4. Emergency Core Cooling

Performance of emergency core cooling systems following an accident may be verified by observing redundant and independent indications as described in Subsection 7.5.1.2 item c and fully satisfies the need for operator verification of operation of the system.

5. Postaccident Tracking

The various indications described in Subsection 7.5.1.2 provide adequate information regarding the status of the reactor vessel level and pressure to allow operators to make proper decisions regarding core and containment cooling operations; they also fully satisfy the need for postaccident surveillance of these variables.

c. Safe Shutdown Display

SEE INSERT 2

~~The applicable instrumentation comes from the list in Subsection 7.5.1.2 and includes the control rod status lamps, scram pilot valve status lamps, and neutron monitoring instrumentation. These displays are expected to remain operable for a long enough time following an accident to indicate the occurrence of safe shutdown.~~

~~Redundancy is provided by dividing the displays into three separate systems. The rod position and neutron monitoring outputs are recorded, the former by the process computer. The systems cited are either manually or automatically connectable to the standby a-c power.~~

INSERT 2

INSERT FOR SAFE SHUTDOWN DISPLAY - P.7.5-11 PARA.7.5.2.2.C

7.5.2.2.C SAFE SHUTDOWN DISPLAY

The applicable instrumentation comes from the list in subsection 7.5.1.2 and includes the computer printout of rod position information, scram pilot valve status lamps and neutron monitoring instrumentation. These displays are expected to remain operable for a long enough time following an accident or loss of off-site power to indicate the occurance of safe shutdown.

Redundancy is provided by dividing the displays into three separate systems. The rod position and neutron monitoring outputs are recorded, the former by process computer. Although the rod position indication on panel P6.3 is lost due to a loss of off-site power, the RPIS cabinet will provide indication of rod position to the computer. (RPIS is powered from the uninterrupted power supply.)

Compliance with IEEE 279-1971

Neither the rod status circuitry nor the scram pilot valve status circuitry alone meets the requirements of IEEE 279-1971. Jointly, however, they do meet the requirements applicable to display instrumentation except for seismic qualification.

The neutron monitoring system is designed to meet all the requirements of IEEE 279-1971 as a part of the reactor protection system. However, its RPS function is a "fail-safe" function, while safe shutdown display is not. Further, its RPS function terminates with the generation and maintenance of a shutdown signal. ~~In this regard, post-DBA environment conditions may cause malfunction but not until the RPS function of scram generation is concluded. This makes it impossible to claim continuous indicating capability for safe shutdown display by the neutron monitoring system. In addition, loss of power is acceptable to the RPS performance, but not to the safe shutdown display performance although such a condition is highly unlikely.~~ Redundancy, power switching capabilities, RPS capabilities, and expected time to failure under DBA environment conditions allow the neutron monitoring system to meet the functional requirements of IEEE 279-1971 as applicable to display instrumentation.

DELETE

Q031.281 <u>(Response)</u>

(1) Section 7.5 has been revised to show that <u>all</u> post-accident instrumentation and read-outs are powered from safety-grade power sources, and the read-out devices themselves, as well as the instrumentation which provides the signals are seismically qualified, in accordance with IEEE 344-1975.

(2) Section 7.5.1.2 has been revised to show that suppression pool temperature <u>is</u> recorded on redundant, seismically-qualified recorders, and that all post-accident tracking parameters are redundant, with at least one channel recorded, and indication on all channels.

(3) Section 7.5.1.2 a) has been revised to show that the rod position information system, which feeds shutdown position information to the process computer is fed from the uninterruptible power supply, and that the control rod scram pilot valve status lamps, and the neutron monitoring power range channels are ultimately fed from safety grade power supplies.

ENCLOSURE 6

ROD BLOCK MONITOR SYSTEM

The NRC Staff has conducted a thorough review of the LaSalle County Station Reactor Manual Control System (RMCS) and specifically the Rod Block Monitor (RBM) Subsystem. This system, the logic and design of which is identical to the Zimmer Station system reviewed in NUREG-0528, represents a design improvement over that previously licensed on the Hatch-2 docket. In that regard, the LaSalle County system design is the same as Hatch-2 except for the new electronic multiplexing feature.

A GE/NRC generic meeting was held on January 26, 1981 to discuss the RMCS and to specifically discuss the propriety of utilizing the RBM in the analysis of certain operational transients. The new electronic RMCS utilized at LaSalle County was described in detail with emphasis on the reliability redundancy and self-testing features of the system. At that time the NRC Staff indicated tentative approval of the design and transient analysis with the possible addition of periodic Technical Specification surveillance to better assure proper system operation. The LaSalle County technical specification will be amended to include this additional functional testing of the multiplexing feature of the RMCS. Proposed additions to the appropriate technical specification are included for information as an attachment to this enclosure

TABLE 4.3.6-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	NA	S/U[b], [redacted]	Q	1*
b. Inoperative	NA	S/U[b], [redacted]	NA	1*
c. Downscale	NA	S/U[b], [redacted]	Q	1*
2. APRM				
a. Flow Biased Simulated Thermal Power - Upscale	NA	S/U[b], M	Q	1
b. Inoperative	NA	S/U[b], M	NA	1, 2, 5
c. Downscale	NA	S/U[b], M	Q	1
d. Neutron Flux - Upscale, Startup	NA	S/U[b], M	Q	2, 5
3. SOURCE RANGE MONITORS				
a. Detector not full in	NA	S/U[b], W[c]	NA	2, 5
b. Upscale	NA	S/U[b], W[c]	Q	2, 5
c. Inoperative	NA	S/U[b], W[c]	NA	2, 5
d. Downscale	NA	S/U[b], W[c]	Q	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U[b], W[c]	NA	2, 5
b. Upscale	NA	S/U[b], W[c]	Q	2, 5
c. Inoperative	NA	S/U[b], W[c]	NA	2, 5
d. Downscale	NA	S/U[b], W[c]	Q	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	NA	Q	R	1, 2, 5**
b. Scram Trip Bypassed	NA	M	NA	1, 2, 5**
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	NA	S/U[b], M	Q	1
b. Inoperative	NA	S/U[b], M	NA	1
c. (Comparator) (Downscale)	NA	S/U[b], M	Q	1

Tech Spec Revision

TECH SPEC REVISION

ATTACHMENT 1 — 4R

TABLE 4.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES:

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.

b. Within 24 hours prior to startup, if not performed within the previous 7 days.

c. When making an unscheduled change from OPERATIONAL CONDITION 1 to OPERATIONAL CONDITION 2, perform the required surveillance within 12 hours after entering OPERATIONAL CONDITION 2.

* With THERMAL POWER ≥ (20)% of RATED THERMAL POWER.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

[redacted]