LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
LaSalle County Station Unit 2	0 5 0 0 0 3 7 4	1 of 1 0

Title (4) Reactor Core Isolation Cooling Steam Line Differential Pressure High Division II Isolation Due to Procedural Inadequacy

Event Date (5)			LER Number (6)			Report Date (7)			Other Facilities Involved (8)	
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Names	Docket Number(s)
										0 5 0 0 0
0 9	0 1	8 8	8 8	0 1 1	0 0	0 9	2 9	8 8		0 5 0 0 0

OPERATING MODE (9): 3

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

__ 20.402(b)	__ 20.405(c)	X 50.73(a)(2)(iv)	__ 73.71(b)
__ 20.405(a)(1)(i)	__ 50.36(c)(1)	__ 50.73(a)(2)(v)	__ 73.71(c)
__ 20.405(a)(1)(ii)	__ 50.36(c)(2)	__ 50.73(a)(2)(vii)	__ Other (Specify in Abstract below and in Text)
__ 20.405(a)(1)(iii)	__ 50.73(a)(2)(i)	__ 50.73(a)(2)(viii)(A)	
__ 20.405(a)(1)(iv)	__ 50.73(a)(2)(ii)	__ 50.73(a)(2)(viii)(B)	
__ 20.405(a)(1)(v)	__ 50.73(a)(2)(iii)	__ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER	
	AREA CODE	
Alan J. McLaughlin, Technical Staff Engineer, extension 576	8 1 5	3 5 7 - 6 7 6 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
D	B N			N					
A				N					

SUPPLEMENTAL REPORT EXPECTED (14)

		Expected Submission Date (15)	Month	Day	Year
__ Yes (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e, approximately fifteen single-space typewritten lines) (16)

On September 1, 1988 at 1631 hours, a Division II Reactor Core Isolation Cooling (RCIC) Steam Line Differential Pressure High Inboard Isolation occurred. Unit 2 was in Hot Shutdown at 100 psig and 340 degrees Fahrenheit.

At the time of this isolation it was misinterpreted as an RCIC Steam Supply Pressure Low Isolation. Instrument Maintenance Department troubleshooting initiated on September 2, 1988 at 0200 hours, upon correct interpretation of the isolation, revealed that pressure differential switch PDS-2E31-N007BB was isolated from the system at its Intermediate Rack Root Valves. A late NRC red phone notification (10CFR50.72) was made following proper classification of the event at 0305 hours on September 2, 1988.

The improper valving of this pressure differential switch occurred on August 17 and August 18, 1988, when this switch had been installed to replace a Static-O-Ring switch which had been found to have a failed diaphragm.

Due to a procedural deficiency, the Intermediate Rack Roots Valves were not verified open prior to returning the switch to service on August 18, 1988.

As reactor pressure was below 150 psig at the time of this spurious isolation, RCIC was not required to be operable during this event per Technical Specification 3.7.3.

This event is reported to the Nuclear Regulatory Commission as a Licensee Event Report in accordance with 10CFR50.73(a)(2)(iv) due to an actuation of an Engineered Safety Feature system.

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PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].

A. CONDITION PRIOR TO EVENT

Unit(s): 2 Event Date: 9/1/88 Event Time: 1631 Hours

Reactor Mode(s): 3 Mode(s) Name: Hot Shutdown Power Level(s): 0%

B. DESCRIPTION OF EVENT

August 17, 1988 DPS-2E31-N007BB Diaphragm Rupture Discovery:

On August 17, 1988, at 1030 hours, with Unit 2 in Mode 1 (Run) at 77% power, the Instrument Maintenance Department began the routine performance of the LaSalle Instrument Surveillance, LIS-RI-401, "Unit 2 Steamline High Flow RCIC Isolation Functional Test," for the 2E31-N007 switches. Per the procedural prerequisites, the Reactor Core Isolation Cooling (RCIC, RI) [BN] system isolation valves were closed and the RCIC system was declared inoperable in accordance with Technical Specifications.

During this surveillance, each of the differential pressure (dP) switches are separately equalized, isolated and pressurized to verify switch closures and the proper isolation signals and alarms occur.

After the 2E31-N007AA, AB and BA switches were satisfactorily tested and returned to service, the 2E31-N007BB switch high instrument isolation valve was closed, the equalizing valve was opened, the low instrument isolation valve closed and then the equalizing valve was reclosed. (See attached Figure 2 of the 2E31-N007BA and 7BB instrument configuration.)

The next step of the surveillance called for applying a differential pressure to the switch until actuation occurs. At this time it was discovered that the switch would not hold any differential pressure. The switch would not actuate when the attempt was made to pressurize, and in addition the test supply water was passing from the Hi side of the switch to the Lo side.

Next, an attempt was planned to determine if the actual switch diaphragm was ruptured or if the 3 valve manifold equalizing valve was leaking. As a Work Request was outstanding indicating that a valve or valves on the manifold was leaking, the Instrument Mechanic proceeded to isolate the 3 valve manifold. In order to accomplish complete isolation of the N007BB switch (and its 3 valve manifold) without affecting the N007BA switch, the Instrument Mechanic closed the Intermediate Rack Root Valves. (See attached Figure 2.)

B. DESCRIPTION OF EVENT (Continued)

Once the N007BB switch was fully isolated, the Instrument Mechanic proceeded to troubleshoot and verified that the diaphragm was indeed ruptured.

Knowing that the switch was to be declared inoperable and replaced, the Instrument Mechanic left the Intermediate Rack Root Valves closed. Also, the 3 valve manifold was equalized and isolated.

LaSalle Work Request L83041 was written to replace DPS-2E31-N007BB. Licensee Event Report 374/88-009-00 further describes the event and reports the failed diaphragm discovery in accordance with I.E. Bulletin 86-02, "Static-O-Ring Differential Pressure Switches."

August 18, 1988 Replacement of Failed Switch:

On August 18, 1988 the failed switch was replaced (by a different work crew) with an identical certified switch. The new switch was then calibrated per LIS-RI-201, "Unit 2 Steam Line High Flow RCIC Isolation Calibration."

Since they are not identified in the procedure, the Instrument Mechanics were not aware that the Intermediate Rack Root Valves had been closed. Therefore the switch remained isolated from the system in an unactuated state.

The new 2E31-N007BB switch remained in this isolated and unactuated state until September 1, 1988 at 1631 hours.

September 1, 1988 Spurious Inboard Isolation of RCIC Steam Line:

On September 1, 1988, at 1631 hours with Unit 2 in Mode 3 (Hot Shutdown) at 100 psig and 340 degrees Fahrenheit (F), DPS-2E31-N007BB actuated causing the following system responses:

1. Auto Closure signal to the RCIC Steam Line Inboard Isolation Valve, 2E51-F063. (Valve went to Closed position.)

2. Auto Closure signal to the RCIC Steam Line Warmup Valve, 2E51-F076. (Valve was in Normally Closed position and remained closed.)

3. "RCIC CHAN B STM LINE DIFF PRESS HI" (B309), alarm window on panel 2H13-P601. Also, the same annunciation occurred on alarm typer.

4. "RCIC DIV 2 ISOL SIG" (B409), alarm window on panel 2H13-P601. Also, the same annunciation occurred on alarm typer.

5. RCIC Turbine Trip Signal. (RCIC Turbine is not required below 150 psig reactor pressure and was not operating.

B. DESCRIPTION OF EVENT (Continued)

In response to the above alarm signals, the Unit Operator acknowledged the alarms at the 2H13-P601 panel. However, the Operator did not recognize this event to be an RCIC Steam Line High dP switch actuation, instead believing that the isolation was due to an RCIC Low Steam Supply Pressure Isolation (which occurs at greater than or equal to 57 psig). The RCIC system was not required to be operable at this time per the Technical Specifications. Had this been a Low Supply Pressure isolation, the "RCIC DIV 1 (or 2) ISOL SIG" and "RCIC STM SUPLY CH A (or B) PRESS LO" alarms would have occurred.

Due to this misunderstanding of the isolation and failure to follow appropriate procedures, no other actions were taken at this time by the Operator.

Later Discovery of Error During Shift Engineer Panel Walkdown:

On September 2, 1988 at 0200 hours, the error in interpreting the annunciator signals and RCIC isolation signal was recognized by the Shift 1 (2300 hours to 0700 hours) Shift Engineer during a panel walkdown.

An actuation of PDS-2E31-N007BA, PDS-2E31-N007BB, PDS-2E31-N013BA, or PDS-2E31-N013BB will provide the identical isolations and alarms, a determination had to first be made as to which switch or switches was actuating. The 2E31-N013BA and 13BB switches are located on flow elbows downstream of the RCIC steam line outboard isolation valve, 2E51-F008. (See Figure 1.)

The Instrument Maintenance Department was immediately called out to troubleshoot the problem. A late red phone notification (4 hour reportable) was made on September 2, 1988, at 0305 hours, due to the Engineered Safety Feature (ESF) actuation which had occurred at 1631 hours on September 1, 1988.

September 2, 1988 Instrument Maintenance Department Troubleshooting:

Through electrical checks on the relays associated with the switches in question, it was determined that PDS-2E31-N007BA and/or PDS-2E31-N007BB was causing the actuations.

At this time a calibration check was begun on switches 2E31-N007BA and 2E31-N007BB, per LIS-RI-201. When the 2E31-N007BA switch was equalized by opening of the instrument equalizing valve the alarm signals remained present in the Control Room. This implied that the 2E31-N007BB switch was tripped, causing the actuations. The 2E31-N007BA switch was returned to service.

The 2E31-N007BB switch was then equalized, at which time the alarms cleared. The calibration was then completed and the 2E31-N007BB switch was found to be slightly below calibration tolerance. The 2E31-N007BB switch diaphragm was evaluated as being intact. The instrument was then satisfactorily recalibrated within allowable tolerances.

The decision was then made to backfill the sensing lines associated with the switch.

B. DESCRIPTION OF EVENT (Continued)

However, after completing the calibration procedure on the 2E31-N007BB switch the Instrument Mechanics performing the surveillance discovered that the two Intermediate Rack Root Valves were in the fully closed position, isolating PDS-2E31-N007BB.

This RCIC Steam Line Differential Pressure High Auto-isolation is reported to the Nuclear Regulatory Commission in accordance with 10CFR50.73(a)(2)(iv) due to an ESF actuation.

C. APPARENT CAUSE OF EVENT

The cause of this event is attributed to procedural deficiency and personnel error.

The LaSalle Instrument Department surveillance and procedures did not address the Intermediate Rack Root Valves and verification that instruments are fully unisolated following calibration or replacement. These Intermediate Rack Root Valves are installed in a limited number of locations.

Personnel error, on the part of Operating personnel, resulted in the RCIC Steam Line High dP isolation being mistaken for a low reactor pressure RCIC isolation. The Operators were placing "A" Residual Heat Removal (RHR) [BO] loop into Shutdown Cooling at the time of the event and saw the RCIC isolation occur. Knowing that RCIC was not required (per Technical Specifications) at less than 150 psig reactor pressure, and expecting an RCIC Turbine isolation and trip signal at 57 psig, they did not recognize that a Differential Pressure Hi isolation had occurred.

Due to the subsequent failure to follow procedures, the situation existed for the remainder of shift 3 (1500 to 2300 hours) and 3 hours into shift 1, prior to being recognized.

Because of the fact that the 2E31-N007BB switch actuated, even though it was isolated, it is postulated that one or both of the Intermediate Rack Root Valves were leaking. Even a very small leak would allow one side to depressurize faster than the other.

Apparently the degree and conditions under which the individual valves leaked (and the direction in which they leaked) allowed the instrument to remain untripped for the time period between August 18, 1988 and September 1, 1988. Then on September 1, 1988, when reactor pressure decreased sufficiently the 2E31-N007BB experienced a high enough dP to actuate at its setpoint.

During normal plant operations, both sides of the differential switch would be at about 990 psig, with this pressure trapped between the Intermediate Rack Root Valves and the diaphragm. After the Unit 2 shutdown on September 1, 1988, reactor pressure was reduced to approximately 100 psig. It would take only a minute amount of leakage to cause the 990 psig in the Lo side instrument leg to drop by the 75.4 inches of water column ("W.C.) setpoint of the switch.

D. SAFETY ANALYSIS OF EVENT

Instrument Design/Functional Description:

Reactor Core Isolation Cooling has four leak detection Pressure Differential Switches, 2E31-N007AA/AB/BA/BB, designed to sense a line break on the Residual Heat Removal Steam Condensing line or the instrument lines themselves.

The switches are connected to instrument lines attached to flow elbows upstream of the RCIC steam line inboard isolation valve, 2E51-F063, and its parallel warmup line isolation valve, 2E51-F076. (See attached Figure 1.)

In the event of a Residual Heat Removal System steam condensing line break downstream of the 2E51-F064 and 2E51-F091 valves (primary containment boundary) both the 2E31-N007AA and 2E31-N007BA switches should close, with the 2E31-N007AA (DIV 1) switch (off of upstream flow elbow instrument lies) isolating the outboard isolation valves F064 and F091. The 2E31-N007BB (DIV 2) switch (off of the downstream flow elbow instrument lines) isolates the inboard RCIC steam line valves, 2E51-F063 and 2E51-F076.

The 2E31-N007AB switch (connected in reverse parallel with 7AA) and 2E31-N007BB switch (connected in reverse parallel with 7BA) will only actuate if their respective Hi pressure instrument line were to break. In the event of a Lo pressure instrument line break the 2E31-N007AA and 2E31-N007BA switches would actuate, closing their respective divisional isolation valves. These instrument lines contain excess flow check valves designed to minimize potential leakage.

Upon any one of the 2E31-N007 switches actuating, a "Steam Line High Flow RCIC Isolation" will occur and the respective division's "RCIC CHAN A (or B) STM LINE DIFF PRESS HI" and "RCIC DIV 1 (or 2) ISOL SIG" alarms would occur. Also, an RCIC Turbine Trip signal would initiate.

The safety consequences of the above delineated events is as follows:

1. The August 18, 1988 failure to return new pressure differential switch PDS-2E31-N007BB to system lineup, left this switch out-of-service. However, as this switch only functions to sense a high side instrument line break, and then initiates closure of the downstream isolation valve, it has a very limited usefulness and its unavailability would not be a great concern. An evaluation of this switch's usefulness and possible removal from system is to be addressed as a corrective action.

2. The procedural inadequacy which led to leaving this switch isolated, is a potentially significant concern and is addressed as such in the corrective actions to this event.

3. The September 1, 1988 actuation of this isolated switch had limited safety consequences under the conditions which is occurred (i.e., Hot Shutdown with unit at 100 psig and 340 degrees F). The Technical Specifications do not require RCIC to be operable below 150 psig.

D. SAFETY ANALYSIS OF EVENT (Continued)

4. While the safety consequences of this spurious actuation would have been considerably greater had the unit been in Run Mode or in an accident condition with RCIC in use, it should be noted that the High Pressure Core Spray System (HPCS) [BG] was available throughout the time period from August 18 to September 1, 1988.

5. The safety consequences of the misinterpretation of the RCIC Steam Line Differential Pressure High isolation for an RCIC Steam Supply Pressure Low isolation are potentially significant and are addressed as such in the corrective actions for this event. However, it is highly improbable that the personnel involved would have made this error had reactor pressure been in the range (greater than 150 psig) in which RCIC is required operable.

E. CORRECTIVE ACTIONS

1. Walkdown all Unit 2 Safety Related instrument racks which have Intermediate Rack Root Valves and verify the valves are open. This was completed prior to Unit 2 Startup. The Instrument Maintenance Department has lockwired these valves in the open position.

2. Instrument Maintenance Department completed a walkdown of all Unit 1 instrument racks as defined for Unit 2.

3. The existence of these Intermediate Rack Root Valves on certain instrument racks will be specified and addressed in LaSalle instrument surveillances and procedures as determined by Onsite Review. The tracking of this effort will be accomplished by Action Item Record (AIR) 374-200-88-03901.

4. Instrument Maintenance Mechanics reviewed this event and lessons learned in department communications meeting on September 2, 1988. Documented training by Instrument Maintenance Training Coordinator for all Instrument Mechanics will be completed. The tracking of this effort will be accomplished by AIR 374-200-88-03902.

5. Instrument Maintenance Department will review practices and consider methods for identification of instrument valves left off-normal. The tracking of this effort will be accomplished by AIR 374-200-88-03903.

6. This event will be highlighted in the Instrument Maintenance Department's continual training program in connection with LER 374/88-003-00, "Reactor Scram on High APRM Flux Level Due to Personnel Valving Error and Procedural Deficiency." The tracking of this effort will be accomplished by AIR 374-200-88-03904.

7. All licensed Operators will be tailgated on this event. The tracking of this effort will be accomplished by AIR 374-200-88-03905.

E. CORRECTIVE ACTIONS (Continued)

8. All individuals involved on the third shift of September 1, 1988 and the first shift of September 2, 1988, will be counseled on the event.

   Failure to follow procedures:

   LAP-200-3, "Shift Change"
   LAP-1600-2, F.1.o., p., q., s., aa. 1) and aa.2), "Conduct of Operations"
   LOA-B309, "RCIC CHAN B STM LINE DIFF PRESS HI"
   LOA-B409, "RCIC DIV 2 ISOL SIG"

   and the failure to notify supervisors of a perceived abnormal condition (57 psig isolation coming too soon), will be emphasized. AIR 374-200-88-03906 will track completion of this item.

9. Disciplinary actions were considered and appropriate actions taken.

10. Shift turnovers will be monitored to address possible improvements. All reviews will be evaluated and possible recommendations for improvement will be made. The tracking of this effort will be accomplished by AIR 374-200-88-03907.

11. LAP-1600-2, "Conduct of Operations," will be revised to clarify proper annunciator response. AIR 374-200-88-03908 will track the completion of this procedure revision.

12. An evaluation has been conducted on the necessity of having these secondary instrument line break isolation switches. Commonwealth Edison Boiler Water Reactor Engineering Department has determined that they are not required. The tracking of further actions with regards to these switches and modifications will be accomplished by Action Item Record 374-200-88-03909.

F. PREVIOUS EVENTS

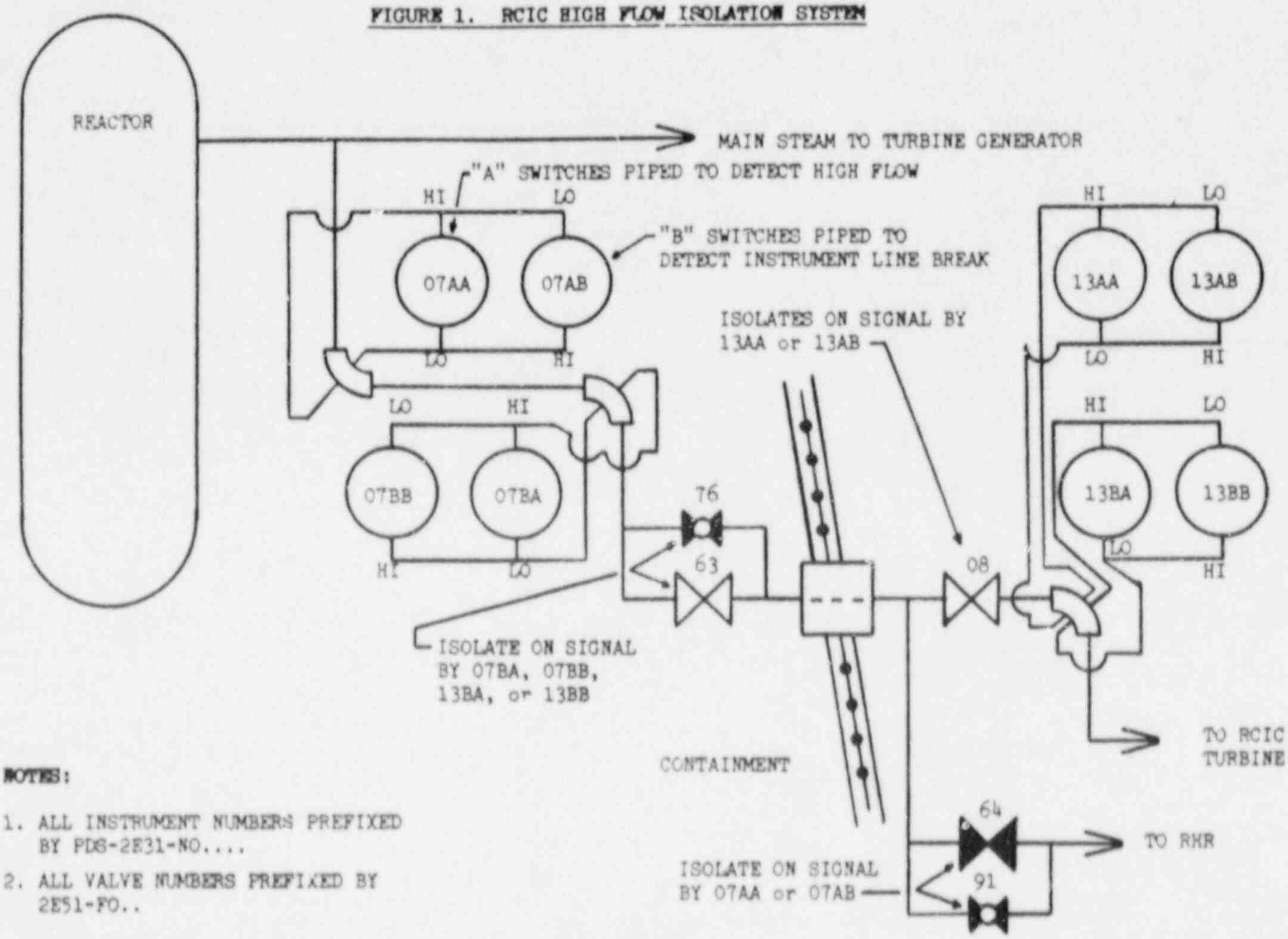
LER Number	Title
373/87-017-00	Drywell Pressure Switch Valved Out Longer than Allowed
374/88-003-00	Reactor Scram on High APRM Flux Level Due to Personnel Valving Error and Procedural Deficiency

G. COMPONENT FAILURE DATA

None.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

FIGURE 1. RCIC HIGH FLOW ISOLATION SYSTEM



NOTES:

1. ALL INSTRUMENT NUMBERS PREFIXED BY PDS-2E31-NO....
2. ALL VALVE NUMBERS PREFIXED BY 2E51-FO..

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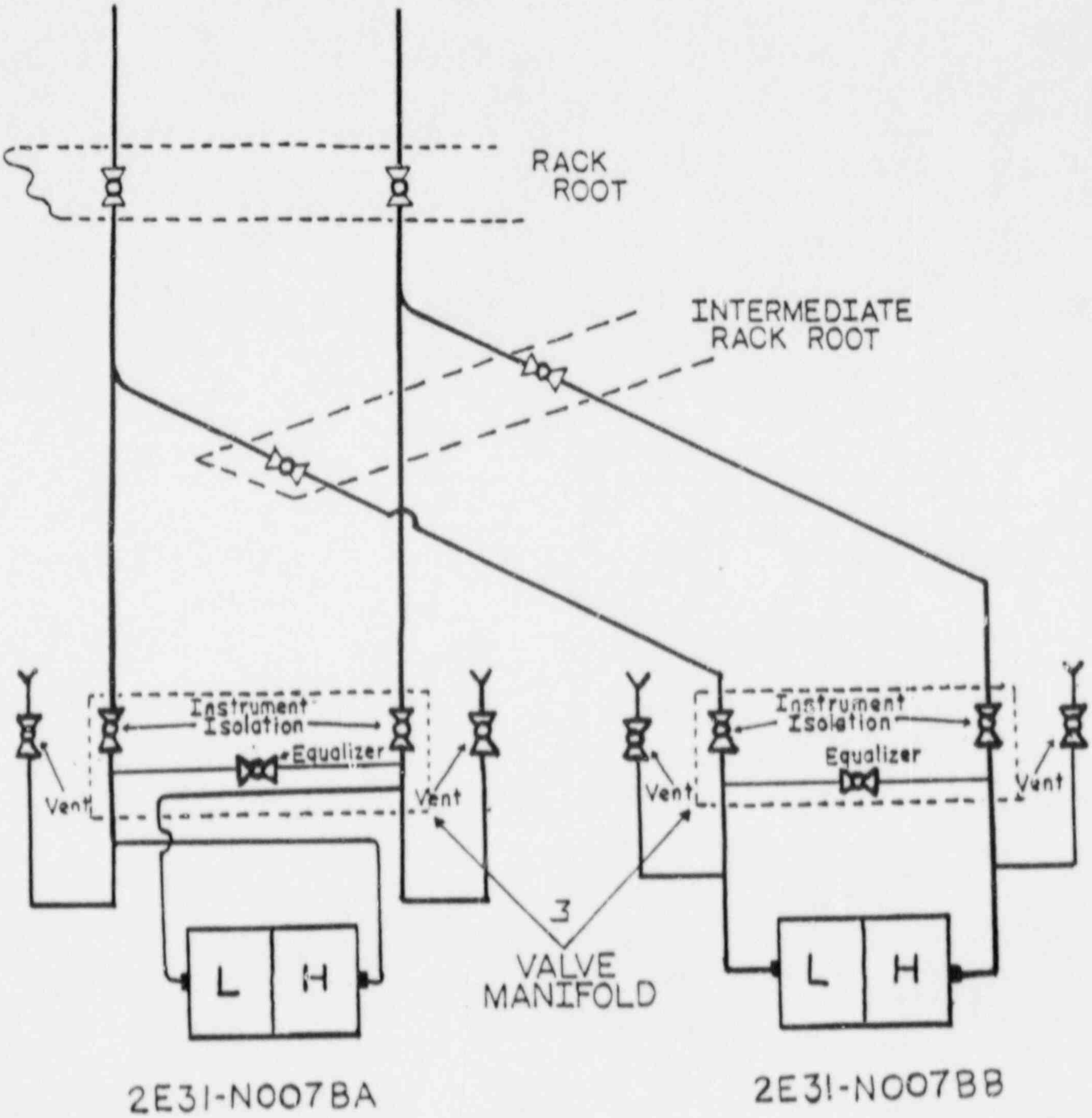
TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

FIGURE 2.



Commonwealth Edison
LaSalle County Nuclear Station
Rural Route #1, Box 220
Marseilles, Illinois 61341
Telephone 815/357-6761

September 29, 1988

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Station P1-137
Washington, D.C. 20555

Dear Sir:

Licensee Event Report #88-011-00, Docket #050-374 is being submitted to your office in accordance with 10CFR50.73(a)(2)(iv).

G. J. Diederich
G. J. Diederich
Station Manager
LaSalle County Station

GJD/AJM/kg

Enclosure

xc: Nuclear Licensing Administrator
NRC Resident Inspector
NRC Region III Administrator
INPO - Records Center

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