Commonwealth Edison
LaSalle County Nuclear Station
2601 N. 21st. Rd.
Marseilles, Illinois 61341
Telephone 815/357-6761

March 12, 1993

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

Dear Sir:

Licensee Event Report #93-004-00, Docket #050-373 is being submitted to your office in accordance with 10CFR50.73(a)(2)(v) and 50.73(a)(2)(i).

Gary Spedl

G. F. Spedl
Station Manager
LaSalle County Station

GFS/db/mkl

Enclosure

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

170097

JE22

LICENSEE EVENT REPORT (LER)

Form Rev 2.0

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

Title (4) Unit 1 Reactor Core Isolation Cooling Inoperable Due To The Inboard Isolation Valve Failure Due To A Breaker Trip On Thermal Overloads

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 2	1 0	9 3	9 3	0 0 4	0 0	0 3	1 2	9 3		0 5 0 0 0

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)
POWER LEVEL (10)	0 8 5	

	20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
	20.405(a)(1)(i)		50.36(c)(1)	X	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)	X	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	
Dave Button, Engineering Motor Operated Valve Group, Extension 2460	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	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	B N		L 2 0 0	Y					

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 February 10, 1993 Unit 1 was in operational condition 1 (Run) at 85% power following the fifth refueling outage. At 1650 hours, while attempting to open the Reactor Core Isolation Cooling System Inboard Isolation Valve 1E51-F063 its circuit breaker tripped on thermal overloads. This occurred during restoration of the RCIC System after preplanned preventative maintenance.

The RCIC Steam Line was isolated for planned maintenance which required the Inboard Steam Isolation Valve 1E51-F063 to be closed. When returning RCIC to normal, the 1E51-F063 breaker tripped on thermal overload when an open signal was sent to the valve.

The RCIC System was declared inoperable and containment integrity was maintained by closing the RCIC Outboard Isolation Valves 1E51-F008 and 1E51-F064 and taking them out of service. Unit 1 shutdown was initiated at 2000 hours on 02/13/93.

Maintenance Departments performed repairs to the valve and found a broken disc retainer pin. This was replaced, the valve was tested, and RCIC was declared operable.

A Generating Station Emergency Plan (GSEP) Unusual Event was declared at 2000 hours on 2/13/93 because of a unit shutdown due to 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)(v) due to RCIC being declared inoperable (loss of a safety system function) and 10CFR50.73(a)(2)(i) due to the completion of a nuclear plant shutdown required by Technical Specification.

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): 1 Event Date: 2/10/93 Event Time: 1650 Hours

Reactor Mode(s): 1 Mode(s) Name: Run Power Level(s): 85%

B. DESCRIPTION OF EVENT

On February 10,1993, at 1650 hours, with Unit 1 in Mode 1 (Run) at 85% power while restoring the Reactor Core Isolation Cooling (RCIC, RI) [BN] System to Standby after pre-planned preventative maintenance, the Nuclear Station Operator (NSO) attempted to open the RCIC Steam Line Inboard Isolation Valve 1E51-F063. The NSO gave the valve an open signal and obtained dual light indication. Approximately 6 seconds later indication was lost due to a thermal overload trip. The RCIC Steam Line Outboard Isolation Valves (1E51-F008 and 1E51-F064) were taken out of service closed as required by Technical Specification 3/4.6.3 for primary containment integrity. The RCIC system was declared inoperable in accordance with Technical Specification 3/4.7.3.

As required for the pre-planned maintenance, the valve was closed and remained closed for approximately 10 hours. During this time no abnormal indications were noted. Upon completion of RCIC System maintenance, the RCIC Steam Line Warm-up Valve 1E51-F076 was opened to equalize pressure and temperature around the 1E51-F063 after which an attempt was made to open the 1E51-F063 from the Control Room. Dual indication was received , followed by a thermal overload trip approximately 6 seconds later. The overload was reset and another attempt was made to open the valve, resulting in a second thermal overload trip. The thermal overload was again reset, this time an ammeter/chart recorder was installed on the power leads. The valve was then given a close signal. Motor current was observed to be in excess of 100 amps for 1-2 seconds, at which time the circuit breaker was tripped manually.

Unit 1 was brought to Cold Shutdown to initiate repairs on the RCIC Steam Line Inboard Isolation Valve 1E51-F063. During the shutdown a Generating Station Emergency Plan (GSEP) Unusual Event condition was entered on 02/13/93 at 2000 hours because of a plant shutdown required by Technical Specifications. The GSEP was terminated at 1800 hours on 02/14/93 when the RCIC System was no longer required for Unit 1 operating condition.

Containment entry was made after unit shutdown to perform a preliminary inspection of the valve/actuator for signs of external damage. No visible damage was noted. The local indicator for the Limitorque Operator Spring Compensator Pack (SB design) indicated the spring compensator was fully compressed. The local position indicator (MDPI) showed the position of the valve to be 10-12% open in the as found condition although it was later determined to be fully closed. The limit switch housing cover was removed to perform an inspection of the torque switch and limit switches. The torque switch was found to be over-rotated and the close setting indicator had slid underneath the torque limiter plate thereby

B. DESCRIPTION OF EVENT CONTINUED

raising the close torque switch setting. The torque switch setting screws open and close were found to be loose. Based on no apparent damage to the limit switches and torque switch, the valve actuator was declutched and the valve was manually cycled to try to determine if the valve was internally bound. The valve was easily cycled using the handwheel with no evidence of binding in the valve or actuator. When manually cycling the valve, the local position indicator had been determined to be shifted. The local position indicator would display a valve position greater than 100% when the valve was full open and indicated approximately 20% open when the valve was full closed.

To continue analyzing the valve condition, the torque switch setting screws were adjusted to a setting of 1.00/1.00 (open/close) and the valve was electrically cycled to the open position. When the valve was cycled to the full open position it was found to be stopping about 2 3/4" from the valve backseat. The limitswitch adjustment was checked and the limits were found to be out of adjustment by approximately 20%, this corresponds to the offset that was observed in the local position indicator when the valve was full closed.

The limit switches were properly adjusted and the valve was electrically cycled three times with the torque switch adjusted to a setting of 1.00 /1.00 (open/close) while the valve was monitored by VALVE OPERATION TEST and EVALUATION SYSTEM (VOTES) diagnostic equipment. No abnormalities were noted in the VOTES traces. The torque switch setting was raised to 1.50/1.50 and the valve was again cycled three times with the valve monitored by VOTES equipment, no abnormalities were noted with the VOTES traces and the thrust characteristics were consistent with the previous (prior to failure) VOTES test results.

The compensator spring assembly and stem nut were removed with the actuator in place and inspected for evidence of abnormal wear or indications which might explain the shift in limit switch settings. No wear was noted and adequate spline engagement between the stem nut and drive sleeve was observed.

The actuator was removed from the valve for disassembly and inspection. No damage was noted with the actuator internals. The electrical components were inspected (cartridge assembly, limit switch assembly and the torque switch). The results of this inspection were 1) the torque switch setting screws being loose, 2) the close torque switch setting block being underneath the torque limiter plate, and 3) the torque limiter plate being slightly bent (pushed away from the torque switch face) on the close setting side and installed between 2 (two) flat washers on the torque switch shaft.

The valve internals were inspected, with no visible physical damage to the valve body. There were no obvious gouges in the valve body, sides and/or seat areas. A Non Destructive Examination (NDE) DYE PENETRANT (PT) exam of the valve seating surfaces was performed and a radial indication was found on the upstream seating surface. This was evaluated by engineering and found to be acceptable.

The valve disc pack and stem were also inspected. The disc retainer pin was found to be sheared (apparently the rotation of the valve stem within the upper disc wedge sheared the pin) with the lower disc retainer found inside the valve bonnet. A gap of approximately 1/8 to 3/16 inch was visible from the top of the upper wedge and the bottom of the shouldered portion of the stem. No additional damage or abnormal wear was noted.

B. DESCRIPTION OF EVENT CONTINUED

The valve stem, disc assembly, and the Limitorque actuator were completely replaced. The re-assembled valve was post maintenance tested by performing electrical motor tests (Megger and Winding Resistance readings), VOTES test, limit switch settings verified, in-service leakage test, and seat leakage test. All results were satisfactory.

No other inoperable equipment/systems contributed to this event. No automatic or manual safety system actuations occurred and none were required. No operator actions contributed to the causation or severity of this event.

This event is reported to the Nuclear Regulatory Commission as a Licensee Event Report in accordance with 10CFR50.73(a)(2)(v) due to RCIC being declared inoperable (loss of a safety system function) and 10CFR50.73(a)(2)(i) due to completion of a nuclear plant shutdown required by Technical Specifications.

C. APPARENT CAUSE OF EVENT

On 2/26/93 another event involving the 1E51-F063 valve occured. This event caused a re-evaluation of the apparent cause of this event. All subsequent investigations will be documented under LER 93-007.

D. SAFETY ANALYSIS OF EVENT

The safety consequences of this event were minimal. Primary Containment was maintained by taking the Outboard Isolation Valves 1E51-F008 and 1E51-F064 out of service closed as required by Technical Specification 3/4.6.3. Adequate core cooling was assured by the alternate redundant High Pressure Core Spray (HPCS) system being fully Operable at all times during this event.

E. CORRECTIVE ACTIONS

1. Reviews were conducted of existing Electrical Maintenance (EM) work practices, training, and procedures for adjustment and tightening of torque switch setting screws. Adequate training and guidelines are provided to ensure that torque switch adjusting screws are tightened properly. Interviews were conducted with personnel involved and past performance evaluated. These personnel were determined to be highly reliable and were confident that the torque switch adjustment screws were tight at the completion of testing. In addition, four (4) valves which were worked on during L1R05 were inspected and found no evidence of loose torque switch setpoint screws. These valves were 1E22-F004, 1E22-F012, 1E21-F021, and 1E51-F008. Based on these reviews and that the plant history has not indicated a problem, the loose torque switch setting screw is believed to be an isolated case and no additional corrective action is required.

2. During troubleshooting and root cause analysis, the condition of the torque switch limiter plate was determined to be a potential contributor to the failure. Action Item Record (AIR) 373-180-93-01401 was generated to review existing EM Department procedures to ensure that adequate direction is provided for inspecting and installing limiter plates.

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

E. CORRECTIVE ACTIONS CONTINUED

3. During root cause evaluation, discussions with valve vendor personnel indicated that the design of this valve made thermal binding of the disk assembly when RCIC was isolated very unlikely. However, since the mechanism by which binding of the valve occurred cannot be exactly determined, it is possible that thermal binding may have played some small part in conjunction with the overthrusting of the valve into its closed seat.

   As such, during plant startup, Operations Personnel will attempt to recreate plant conditions by closing MOV 1E51-F063 from rated temperature and pressure in accordance with LaSalle Operating Procedure LOP-RI-03 "Reactor Core Isolation Cooling System Isolation and System Shutdown". After approximately 8 hours, Operations Personnel will warmup and unisolate RCIC in accordance with LaSalle Operating Procedure LOP-RI-05 "Preparation For Standby Operation of The Reactor Core Isolation Cooling System". During valve cycling, EM Personnel will monitor motor current to detect whether any binding of the valve is observed during valve seating and unseating. If increased motor current is observed, further evaluation will be performed to determine if additional warmup requirements are needed prior to operating this valve.

   A Station On-Site Review was conducted, prior to startup, to determine the cause of the failure of the 1E51-F063 valve. Based on the information available at the time, hydraulic locking of the valve was addressed and determined to not be a factor. The associated evaluation stated that an energy source to increase bonnet pressure was not available during the out-of-service interval. Therefore, hydraulic bonnet locking was considered not possible.

4. AIR 373-240-93-00801 has been written for Engineering Personnel to perform a post failure analysis of the wedge retaining pin. This analysis will include materials verification and strength determination of the sheared pin, verification that the stem was fully threaded into the wedge "carrier" prior to retaining pin assembly, analysis of the failure mechanism to help determine under what conditions the pin sheared, and to determine if internal component design changes are appropriate.

5. Further investigation is required due to subsequent problems with the RCIC 1E51-F063 valve. These additional findings will be documented in LaSalle Stations LER 373/93-007.

F. PREVIOUS EVENTS

None.

G. COMPONENT FAILURE DATA

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Anchor Darling	Retaining Wedge Pin	N/A267	267