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

March 26, 1993

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

Dear Sir:

Licensee Event Report #93-007-00, 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/MW/mkl

Enclosure

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

250021

JE22

LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
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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)
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0 2	2 6	9 3	9 3	0 0 7	0 0	0 3	2 6	9 3		0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 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)	___ 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	
M. Wendling, Engineering Motor Operated Valve Group, Extension 2728	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
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 26, 1993 Unit 1 was in operational condition 1 (Run) at 100% power following the fifth refueling outage. At 1223 hours, while attempting to open the Reactor Core Isolation Cooling (RCIC)System Inboard Isolation Valve 1E51-F063 its circuit breaker tripped on thermal overloads. This occurred during testing of the RCIC 1E51-F063 valve.

The RCIC Steam Line testing was being performed to recreate conditions of the previous failure on February 10,1993. 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 02/27/93.

The cause of this event was due to hydraulic lock in the bonnet of the 1E51-F063 valve.

Maintenance Departments performed repairs to the valve and replaced the broken disc retainer pin with a stronger material. Elimination of hydraulic lock was accomplished by the addition of holes drilled into the upstream disc. The valve was tested and RCIC was declared operable.

A Generating Station Emergency Plan (GSEP) Unusual Event was declared at 2000 hours on 2/27/93 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.

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

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: 2/26/93 Event Time: 1223 Hours

Reactor Mode(s): 1 Mode(s) Name: Run Power Level(s): 100%

B. DESCRIPTION OF EVENT

On February 26,1993, at 1223 hours, with Unit 1 in Mode 1 (Run) at 100% power during Reactor Core Isolation Cooling (RCIC, RI) [BN] testing of the 1E51-F063 RCIC Steam Line Inboard Isolation Valve, 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 10 seconds later indication was lost due to a thermal overload trip. This testing was being performed to recreate conditions of the previous failure on February 10, 1993 as part of corrective actions from LER 373/93-004. 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 and a 14 day timeclock was entered in accordance with Technical Specification 3/4.7.3.

This failure occurred while recreating conditions from the initial RCIC valve failure that occurred on 2/10/93. The RCIC 1E51-F063 valve was closed at 0400 on 02/26/93 with no abnormalities noted. At 1200 hours the RCIC 1E51-F008 and 1E51-F076 valves were opened to start warming of the downstream RCIC piping. At 1220 pressure was equalized across the 1E51-F063. Operations attempted to open the 1E51-F063 valve electrically with Electrical Maintenance monitoring motor current. Breaker tripped on thermal overload. Current trace indicated that the disc retaining pin sheared, allowing the stem to back out of the disc partially. The NSO confirmed that RCIC steam supply pressure was decreasing indicating that the 1E51-F063 valve was closed. Operations closed and deenergized the 1E51-F008 valve for primary containment integrity.

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 02/27/93 at 2000 hours because of a plant shutdown required by Technical Specifications. The GSEP was terminated at 1716 hours on 03/01/93 when the RCIC System was no longer required for Unit 1 operating condition.

Containment entry was made after unit power was reduced to about 3% and the drywell deinerted. The test team entered the drywell at 1245 hours on 02/28/93 to perform an inspection of the valve/actuator for signs of external damage. No visible damage was noted. The local position indicator (MDPI) showed the position of the valve to be 36% open in the "as found" condition although it was determined to be fully closed. The limit switch housing cover was removed to perform an inspection of the torque switch and limit switches with no damage found. The test team attempted to operate the handwheel in the closed direction and observed rotation of the valve stem indicating that the stem disc retaining pin had sheared.

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B. DESCRIPTION OF EVENT CONTINUED

The team continued to turn the handwheel in the closed direction in attempt to thread the valve stem back into the disc. Considerable binding was experienced and deflection of the SB Compensator Spring Assembly position was observed as the stem was turned into the disc. The test team stopped turning the handwheel when MDPI indicated 0% and SB compensator indicated 0.47" deflection which was equivalent to the deflection noted during previous testing when the valve was electrically closed.

At 1530 hours Operating Personnel commenced a warmup of the RCIC piping in preparation for attempting to open the 1E51-F063. The 1E51 F008 and 1E51 F076 were opened and after 15-20 minutes the 1E51-F008 was closed. Operations noted that RCIC Drain Pot operation, after 1E51-F076 was opened, indicated that some water had drained when the valve was opened. The test team entered the drywell at 1615 hours and noted the bonnet temperature was approximately 270 degrees and no packing leak observed. The test team attempted to open the 1E51-F063. The valve was manually opened with no abnormal drag or stem rotation. Valve bonnet temperature rapidly rose to 520 degrees Fahrenheit and no packing leakage observed. The valve was then manually closed. The 1E51-F076 valve was close to allow the piping to cool down.

At 1800 hours on 02/28/93, the valves were lined up to recreate the plant conditions existing during the initial valve failure by allowing the valve to remain isolated until about 0500 on 3/01/93. The Technical Staff limited the time the 1E51-F076 was open prior to attempting to open the 1E51-F063.

At 0500 hours on 03/01/93 the test team entered the drywell and observed that the valve bonnet temperature was about 270 degrees Fahrenheit. No packing leakage was observed. Operations opened the 1E51-F076 to commence warming of the RCIC piping. After approximately 15 minutes, the test team observed that the valve bonnet temperature was approximately 300 degrees F, water leaking from the packing gland at a steady trickle, and packing gland nuts tighter than previously observed. Next the team attempted to open the 1E51-F063 manually. The spring compensator was observed to relieve normally and the stem was observed to move about 1/16-1/8 ". The MDPI still indicated 0%. At this point, the handwheel became very difficult to turn requiring 2 people to turn it. The handwheel was operated until the MDPI indicated 3-4% open but no stem movement was observed. The packing nuts were loosened in an attempt to relieve pressure in the bonnet. No movement of the follower was noted, however, packing leakage increased. Additional attempts to open the valve were unsuccessful. After a 15-20 minute period to allow bonnet pressure to bleed off through the packing leakage, the test team attempted to manually open the valve again. Bonnet temperature was approximately 340 degrees F. and packing leakage had decreased. The handwheel was still very difficult to turn and no movement of the stem was noted. The packing nuts were again loosened increasing the packing leakage and some outward movement of the packing follower was noted. The handwheel still would not open the valve. The team elected to have the 1E51-F076 closed in the belief that a slight cooldown of the valve coupled with the packing leakage would relieve bonnet pressure allowing the valve to open. After about 15-20 minutes, the team again attempted to open the valve. Bonnet temperature was approximately 370 degrees F. and the packing leakage had decreased. Additional attempts to open the valve was unsuccessful. The valve handwheel was turned in the closed direction enough to relieve stem tension and the test team exited the drywell.

In conclusion, based on the events as listed above, it was determined by all the data that the valve bonnet became hydraulically locked.

B. DESCRIPTION OF EVENT CONTINUED

An independent review of the 1E51-F063 failure was performed on 3/2/93. Personnel from Nuclear Engineering Department (NED) and Mechanical and Structural Engineering Department (MSD) Departments, Anchor Darling representatives, Impell, LaSalle Station, and the Lead NRC Resident for LaSalle were in attendance. Both failures were described (2/10/93 and 2/26/93) to the panel along with testing performed to identify the problems and the findings upon disassembly of the valve after the first failure. The panel discussed potential failure mechanisms.

Also all the data was presented to the onsite review panel. The onsite review panel evaluated the test data results and determined the cause to be hydraulic locking of the valve. This information is documented in LaSalle Onsite Review 93-010.

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 the completion of a nuclear plant shutdown required by Technical Specifications.

C. APPARENT CAUSE OF EVENT

The apparent cause of the 1E51-F063 valve failure was hydraulic lock. Steam trapped in the bonnet when the valve was closed, condensed as the bonnet cooled creating a lower pressure and allowing high pressure steam/water to enter the bonnet through the upstream disc. The process was then repeated until the bonnet was full of water. After some period of time during which the bonnet temperatures had cooled, steam was admitted to the downstream side by opening the 1E51-F076 bypass valve. This action caused the solid fluid volume in the bonnet to expand and raised pressure significantly above line pressure. This increase in pressure hydraulically locked the valve in the seat.

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 High Pressure Core Spray (HPCS) system being fully Operable at all times during this event.

E. CORRECTIVE ACTIONS

1. The actuator was removed from the valve for disassembly and for inspection. The valve internals were inspected, with no visible physical damage to the valve body. There were no obvious gouges in the valve body, sides, or seat areas. A Non Destructive Examination (NDE) Dye Penetrant (PT) exam of the valve seating surfaces was performed and found to be acceptable. The valve disc pack and stem were also inspected.

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E. CORRECTIVE ACTIONS CONTINUED

2. The Valve Stem, Disc Assembly, and the Limitorque Actuator were completely replaced. The Wedge Retaining Pin was replaced with a stronger material. Two holes were drilled into the upstream disc to eliminate hydraulic locking. The re-assembled valve was Post Maintenance Tested by performing Electrical Motor Tests, Valve Operation Test & Evaluation System (VOTES) Test, Limit Switch Settings verified, In-Service Leakage Test, and Seat Leakage Test. All results were satisfactory.

3. During Root Cause Evaluation, discussions with the Valve Vendor personnel indicated that the design of this valve made thermal binding of the disc assembly when RCIC was isolated very unlikely. However, AIR 373-240-93-01003 was generated to evaluate the need for a thermal analysis on the 1E51-F063 valve.

4. During plant startup, Operations Personnel attempted to recreate plant conditions by closing Motor Operated Valve (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 warmed up the RCIC piping by opening the 1E51-F076 valve. After approximately 1 hour the test team entered the drywell to manually open the 1E51-F063. The Team was successful in manually unseating and opening the valve with no indications of hydraulic locking or any other problem. Operations Personnel then cycled the valve electrically and unisolated the system in accordance with LaSalle Operating Procedure LOP-RI-05 "Preparation For Standby Operation of The Reactor Core Isolation Cooling System".

5. AIR 373-240-93-01002 has been written for Engineering Personnel to perform a post failure analysis of the wedge retaining pin.

6. AIR 373-240-93-01001 has been written to track the evaluation of the operation of the 1E51-F008 for additional administrative controls.

7. AIR 373-455-89-368R1S1.1 has been issued to evaluate incorporation of General Electric Service Information Letter (GE SIL) into Engineering Design practices for the selection of Gate Valves to minimize the potential for Thermal Binding and Pressure Locking.

8. AIR 373-455-89-368R1S1.2 has been issued to re-evaluate the stations original response to GE SIL 368R1S1 titled "Gate Valve Lockup".

F. PREVIOUS EVENTS

A NPRDS (Nuclear Plant Reliability Data System) sort was completed using parameters relative to this event. Three similar failures were identified addressing hydraulic locking.

LER Number	Title
373/93-004-00	Loss of U-1 RCIC
373/83-117	Failure of 1E12-F003B Valve to Open.
373/83-147	Loss of RHR "B" Shutdown Cooling.

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G. COMPONENT FAILURE DATA

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Anchor Darling	Paralled Disk Gate Valve	N/A	

SHEET 7

EVENT SUMMARY AND CAUSE CODES

DVR Number
01-1-93-023

- [ ] Lost generation
- [ ] Cost > $25,000
- [ ] Hazard or Spill
- [ ] Personnel injury
- [ ] Reactor trip
- [ ] ESF actuation
- [ ] NRC reportable
- [x] LER
- [ ] PSR
- [ ] NRC violation, level___
- [ ] GSEP event, class___
- [ ] Tech Spec LCO
- [ ] Potential or future lc
- [ ] SALP functional area.

Component type / Failure mode / Department

	Component type	Failure mode	Department	
X	C M	M S	M M	
X				
X				

Licensed? L or blank / Level / Department / Type / Detail code

	Licensed? L or blank	Level	Department	Type	Detail code	
A						
A						
A						

Type / Detail Code / Department

	Type	Detail Code	Department	
B				
B				
B				

	Type	Detail code	
C			

Type of deficiency / Detail code / Procedure type

	Type of deficiency	Detail code	Procedure type	
D				
D				
D				

Type / Detail code / Department

	Type	Detail code	Department	
E				
E				
E				