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ComEd

December 17, 1997

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Licensee Event Report #97-040-00, Docket #050-373 is being submitted to your office in accordance with 10 CFR 50.73(a)(2)(ii).

Respectfully,

Fred Dacimo
Plant General Manager
LaSalle County Station

Enclosure

cc: A. B. Beach, NRC Region III Administrator
M. P. Huber, NRC Senior Resident Inspector - LaSalle
C. H. Mathews, IDNS Resident Inspector - LaSalle
F. Niziolek, IDNS Senior Reactor Analyst
INPO - Records Center

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NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 05/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
LaSalle County Station, Unit 1	05000373	1 of 4

TITLE (4) Unanalyzed Condition for Alternate Shutdown Cooling Steam Lines Due to Inadequate Implementation of Licensing Basis

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 NAME	DOCKET NUMBER
									LaSalle County Station, Unit 2	05000374
10	24	97	97	040	00	12	17	97	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	4
POWER LEVEL (10)	000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(iii)	☐ 73.71(b)
☐ 20.2203(a)(1)	☐ 20.2003(a)(3)(ii)	☐ 50.73(a)(2)(iv)	☐ 73.71(c)
☐ 20.2203(a)(2)(i)	☐ 20.2003(a)(4)	☐ 50.73(a)(2)(	☐ OTHER
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)	☐ 50.73(a)(2)(vii)	
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(viii)(A)	(Specify in Abstract
☐ 20.2203(a)(2)(iv)	☐ 50.73(a)(2)(i)	☐ 50.73(a)(2)(viii)(B)	below and in Text,
☐ 20.2003(a)(2)(v)	☒ 50.73(a)(2)(ii)	☐ 50.73(a)(2)(x)	NRC Form 366A)

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
T. Best, Design Engineer	(815) 357-6761 Extension 2979

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

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
☒ YES (If yes, complete EXPECTED SUBMISSION DATE)	☐ NO		06	01	98

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

Since plant startup when alternate shutdown cooling flow paths were used per procedure LOP-RH-17, "Alternate Shutdown Cooling", the plant has been in an unanalyzed condition. On October 24, 1997, no piping analysis could be found that evaluated the affected piping during its use as an alternate shutdown cooling method. The unanalyzed condition resulted from Reactor Core Isolation Cooling [BN PRA] (RCIC) and Main Steam [SB] (MS) lines, used for alternate shutdown cooling (abnormal operation), not being analyzed filled with water and addressing seismic Category II/I concerns. These lines are filled with steam during normal operation. The root cause investigation is continuing and appropriate corrective actions and safety analysis will be reported in a LER supplement. A similar recent occurrence was LER 97-039 which involved an unanalyzed condition for Safety Relief Valve (SRV) discharge lines with SRVs temporarily removed during SRV bench testing. This event is reportable per 10 CFR 50.73(a)(2)(ii)(A) as being an unanalyzed condition that could have compromised plant safety.

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): 1/2 Event Date: 10/24/97 Event Time: 1200 Hours
Reactor Mode(s): 4/N Mode(s) Name: Cold Shutdown/Defueled Power Level(s): 0%/0%

B. DESCRIPTION OF EVENT

Both LaSalle Units have been in an extended outage since September 1996 with Unit 1 in cold shutdown and Unit 2 defueled. There were no inoperable components or systems that contributed to the event.

While performing the engineering associated with removing the SRVs for the current Unit 1 forced outage, some lines that during normal operation are filled with steam could potentially be filled with water. These lines therefore needed to be analyzed to address seismic II/I concerns. The Engineer involved remembered performing a similar calculation in April 1997 (Calculation L-001122) in conjunction with the use of LOP-RH-17 "Alternate Shutdown Cooling", but seismic loads were not considered to address seismic II/I concerns. A review of Calculation L-001122 confirmed there was no seismic II/I evaluation performed. This is an unanalyzed condition since these lines are used filled with water per procedure LOP-RH-17.

This procedure has been available for use since February 1984 and has been used or taken credit for many times. For the present forced outage for Unit 1, an operability evaluation was performed since this path was relied on for an alternate decay heat removal path and may have been required if both trains of Residual Heat Removal cooling were lost (Technical Specifications Section 3/4.9.11.2). The evaluation showed there were no operability concerns. This event is reportable per 10 CFR 50.73(a)(2)(ii)(A) as being in an unanalyzed condition that could have compromised plant safety.

The immediate corrective action was to perform the seismic II/I analyses for the affected pipes filled with water. Results for Unit 1 show the piping is adequately supported and pipe stresses are below the ASME Code allowables for faulted conditions. Unit 2 results will be completed prior to startup and submitted in a LER supplement.

The inadequately analyzed Main Steam Line B to RCIC takeoff line was analyzed by the NSSS Vendor, and the RCIC steam lines outside containment was analyzed by the AE for the original design. It is not known if those calculations included water loads. In the late 1980's the AE reanalyzed the MS and RCIC piping systems for normal operating modes (i.e. steam) while performing the LaSalle Snubber Reduction Program.

C. CAUSE OF EVENT

The licensing basis for the use of alternate shutdown cooling does not clearly define the seismicity requirements of the affected systems. However, it is clear that certain systems that could affect other safety related equipment do require a seismic II/I evaluation.

The root cause is under investigation and will be reported in a LER supplement.

Inadequate 10 CFR 50.59 safety evaluations for revisions and development of alternate shutdown cooling procedures are contributing causes.

D. SAFETY ANALYSIS

For Unit 1 there are no safety consequences since the analyses for the RCIC steam line and 'B' MS Line show the stresses to be below the ASME Code allowables for faulted conditions. Analyses for Unit 2 are expected to be the same; however results will be submitted in a supplement. UFSAR sections 5.4.7.2.2.1 and 15.2.9.3 discuss alternate shutdown cooling methods. Review of the stress calculations of record indicates that water filled piping loads were not considered for these UFSAR identified alternate shutdown cooling methods. Also, investigations are continuing for other alternate cooling flow paths to assure adequate bases and calculations exist.

E. CORRECTIVE ACTIONS

The immediate corrective action was to analyze the LOP-RH-17 Unit 1 RCIC flow path including seismic loads.

The corresponding analysis for Unit 2 is in progress (NTS # 373-180-97-SCAQ00040.01).

The licensing basis for alternate shutdown cooling methods will be determined (NTS # 373-180-97-SCAQ00040.02).

The other alternate shutdown cooling methods will also be analyzed (NTS # 373-180-97-SCAQ00040.03).

An LER supplement will be submitted to report the root cause investigation and results of other analyses discussed above (NTS # 373-180-97-SCAQ00040.04).

No corrective action is required for the inadequate 10 CFR 50.59 safety evaluations for procedure preparations and revisions because of improvements made to this process during the current outage and recent higher standards for evaluations.

F. PREVIOUS OCCURRENCES

LER NUMBER	TITLE
373-97-027	Equipment Not Seismically Restrained in Seismic Category I Areas Due To Ineffective Implementation of Requirements.
373-97-039	Unanalyzed Condition While Removing Main Steam Safety Relief Valves Due To Inadequate Procedural Guidance

G. COMPONENT FAILURE DATA

Since no component failure occurred, this section is not applicable.