10 CFR 50.73

PHILADELPHIA ELECTRIC COMPANY

LIMERICK GENERATING STATION
P. O. BOX A
SANATOGA, PENNSYLVANIA 19464

August 25, 1989

Docket No. 50-352
License No. NPF-39

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Licensee Event Report
Limerick Generating Station - Unit 1

This revised LER reports a condition prohibited by Technical Specifications in that pressure, flow, and level indication was inoperable and TS actions were not taken within the required time period due to environmentally unqualified valve seals. This LER also reports a condition that could have prevented the fulfillment of a safety function of systems that are needed to mitigate the consequences of an accident.

Reference:	Docket No. 50-352
Report Number:	01-89-034
Revision Number:	01
Event Date:	May 12, 1989
Report Date:	August 25, 1989
Facility:	Limerick Generating Station P.O. Box A, Sanatoga, PA 19464

This revised LER is being submitted to correct an error in the number of valves reported to be involved in this event and to provide minor clarifications. The original LER was submitted pursuant to the requirements of 10 CFR 50.73(a)(2)(i)(B) and 10 CFR 50.73(a)(2)(v)(D). Changes in this revised LER are indicated by revision markers in the right hand margin.

Very truly yours,

for

M. J. McCormick, Jr.
Plant Manager

WGS:sc

cc: W. T. Russell, Administrator, Region I, USNRC
T. J. Kenny, USNRC Senior Resident Inspector, LGS

IE22
1/1

NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Limerick Generating Station, Unit 1	0 5 0 0 0 3 5 2	1 OF 0 8

TITLE (4) This LER Reports a Condition Prohibited by Technical Specification In That Pressure Level and Flow Indication Were Inoperable Due to an Inadequate Modificaton Review

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	1 2	8 9	8 9	0 3 4	0 1	0 8	2 5	8 9		0 5 0 0 0
										0 5 0 0 0

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

OPERATING MODE (9)	5								
POWER LEVEL (10)	0 0 0	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, NRC Form 366A)
		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)
C. R. Endriss, Regulatory Engineer, Limerick Generating Station	2 1 5 3 2 7 1 - 1 1 2 0 0

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)	X NO				

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

On May 12, 1989, with Unit 1 in a refueling outage, station personnel were notified that certain excess flow check valve test valves would not remain leak tight during post Design Basis Accident (DBA) conditions. Nineteen (19) test valves in various applications were found to be in nonconformance with Regulatory Guide (RG) 1.97 requirements. The test valve materials revealed teflon coated viton seals that formed the pressure boundary between the valve plug and body would breakdown during post DBA conditions causing a loss of pressure boundary and instrumentation. The nineteen (19) valves were removed by May 13, 1989. On May 22, 1989, six (6) of the nineteen test valves were identified by plant staff as required to be operable by R.G. 1.97 and Technical Specifications (TS) Section 3.3.7.5. The consequences of this condition were minimal in that a DBA condition did not occur during the time period that the valves were installed and, therefore, no degradation of the pressure boundary occurred. The cause of the event was a result of inadequate review of system design specifications by plant staff supervision prior to receipt of the Low Power Operating License. This LER reports a condition prohibited by TS and a condition that could have prevented the fulfillment of a safety function of systems that are needed to mitigate the consequences of an accident. This LER is being submitted pursuant to 10 CFR 50.73(a)(2)(i)(B) and 10 CFR 50.73(a)(2)(v)(D).

Plant Conditions Prior to the Event:

Operating Mode: 5 (Refueling)

Reactor Power: 0%

Description of the Event

The Limerick Generating Station Final Safety Analysis Report (FSAR), Chapter 1, "Introduction of General Description of the Plant," Section 1.8, states that the requirements of Regulatory Guide (R.G.) 1.97, Revision 2, 1980, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," are met to the extent discussed in FSAR Section 7.5. R.G. 1.97 identifies the plant variables to be measured and the instrumentation criteria for assuring acceptable emergency response capabilities during and following a Design Basis Accident (DBA).

During June 1988, as the result of a review of surveillance test methods on Unit 2, a request by station personnel for the addition of test valves (EIIS:TV) in Unit 2 instrument lines (similar to Unit 1) was initiated. In response to this request, in December 1988, Philadelphia Electric Company (PECo) personnel found that the one-hundred nine (109) NUPRO brand valves that had been installed on Unit 1 instrument lines during the Unit 1 preoperational testing phase prior to receipt of the Low Power Operating License were non-ASME code valves. Ninety-nine (99) were installed on General Electric (GE) supplied local instrument racks outside the ASME code boundaries and ten (10) were installed on non-GE supplied level instrument racks within the ASME code boundaries. The test valve installations in question were designed to allow simulation of a line break in the instrument lines in order to functionally check the primary containment excess flow check valves.

In January 1989, Philadelphia Electric Company (PECo) requested that the Architect Engineer (A/E) initiate a Modification Design Change Package (MDCP) to revise ASME code boundaries and pipe classifications for Unit 1 that would have allowed the installation of the subject test valves. An Engineering Work

Request (EWR) was initiated in March 1989 to obtain formal engineering resolution of the issue. At the end of April 1989 a Nonconformance Report (NCR) was issued to formally identify the non-ASME test valves on ASME code lines as an item to be resolved. Subsequent investigation then revealed that the Unit 1 test valves had not been procured in accordance with design specifications. On May 6, 1989, a second NCR was issued and formally identified that the test valves were not Environmentally Qualified (Q) for safety related applications but were installed in Q-passive applications (i.e., safety related item, however, its only function is to maintain the required pressure boundary).

On May 12, 1989, the plant staff was notified by the A/E that the ten (10) test valves installed on non-GE supplied instrument racks satisfied the ASME code boundary criteria as defined in ASME Section XI, IWA 7000. For the ninety-nine (99) NUPRO valves installed on General Electric rack mounted instruments, the valves were originally installed beyond the ASME boundaries and therefore the ASME Code does not apply to these valves. However, an environmental qualification problem was identified with the soft parts of the NUPRO valves. The test valves contained Teflon coated Viton seals (EIIS:SEAL) as part of the valves pressure boundary. The A/E calculated that the radiation dose that would be received by the test valves eight (8) hours after a DBA Loss of Coolant Accident (LOCA) would cause degradation of the valve seals leading to the potential to allow leakage causing inaccurate instrument indications. As a result of the potential for the loss of the required instrumentation due to breakdown of the teflon coated viton seals, nineteen (19) test valve installations were removed on May 13, 1989. The test valves had been installed on various instrument lines and fifteen (15) of nineteen (19) test valves were installed on instrumentation for vessel level (EIIS:LI) and pressure indications (EIIS:PI), Main Steam Line Isolation Valve (MSIV) Leakage Control (EIIS:BD) system pressure indications, and reactor recirculation pump (EIIS:P) flow indications (EIIS:FI). These indications are required to be operable longer than eight (8) hours following a DBA LOCA to comply with R.G. 1.97. The purpose of this instrumentation is to assure acceptable emergency response capabilities during and following the course of an accident. In addition, the remaining four (4) of the nineteen (19) test valves associated with the Reactor Core Isolation (RCIC)(EIIS:BN) system steam supply instrumentation were also replaced as a conservative measure. This instrumentation is only required to be operable

for one (1) hour following the DBA LOCA in accordance with R.G. 1.97 and the instrumentation would be unaffected by the condition of the test valve seals.

On May 22, 1989, plant staff identified that six (6) of the nineteen (19) test valves were also located on instrument lines that are connected to instrumentation required to be operable by R.G. 1.97 and Technical Specifications (TS). The environmentally unqualified test valves affected the operability of these six (6) instruments and TS Limiting Condition for operation (LCO) 3.3.7.5 requires that a minimum number of one channel of reactor vessel pressure and level instrumentation be operable. However, this TS was not met due to the environmentally unqualified test valve seals and the required action was not taken.

This condition has existed since October 26, 1984, the date of the issuance of the Unit 1 Low Power Operating License. The "Action" required by TS LCO 3.3.7.5 was not taken in the specified time period, therefore, for the six (6) test valves associated with the TS required instrumentation, this constitutes a condition prohibited by TS and is reportable in accordance with 10CFR 50.73(a)(2)(i)(B). Additionally, for the fifteen (15) test valves associated with the instrumentation required by R.G. 1.97, this constitutes a condition that could have prevented the fulfillment of a safety function of systems that are needed to mitigate the consequences of an accident and is reportable in accordance with 10CFR50.73(a)(2)(v)(D).

Accordingly, a four-hour notification to the NRC was made on May 24, 1989 at 1615 hours in accordance with the requirements of 10CFR 50.72(b)(2)(iii)(D).

Consequences of the Event:

The actual consequences of this condition were minimal because a DBA LOCA that could result in degradation of the test valve pressure boundary seal did not occur and the installation of these valves does not impact normal plant operations. There was no release of radioactive material to the environment as a result of this condition.

The potential consequences of this condition were that instrumentation for Reactor level and pressure, MSIV Leakage Control system pressure and Reactor Recirculation pump flow may not have provided the accurate indications necessary to assess equipment and plant conditions following a DBA LOCA. The Control Room operator would utilize the post accident qualified instrument indications to monitor transient reactor plant behavior and to verify proper safety system performance following the accident. Approximately eight (8) hours following the DBA LOCA, the test valve seals may start to leak due to radiation exposure degradation, therefore, making post accident qualified instruments inaccurate. The inaccurate indications could affect the ability of the Control Room operators to verify adequate reactor coolant inventory and pressure, adequate MSIV Leakage Control system operation and Reactor Recirculation pump flow.

The leaking test valve seals would have also resulted in contamination of certain areas of the Reactor Enclosure (EIIS:NG). However, personnel access to the Reactor Enclosure would be restricted after the DBA LOCA due to expected radiation levels resulting from the accident regardless of the condition of the test valve seals. The contaminated fluid that leaked from the valves would be processed by the floor drain system and the Reactor Enclosure Recirculation system (EIIS:VA) (RERS) and Standby Gas Treatment system (EIIS:BH) (SGTS) would process the airborne contamination.

If a DBA LOCA had occurred, the radiation induced degradation of the NUPRO valve seals and resultant leakage could have adversely affected the ability of the plant and Control Room operators to mitigate the consequences of the accident.

Cause of the Event:

This event was a result of an inadequate review of those system design specifications against the materials (i.e. valves) intended for installation on the part of plant staff supervision. The valves located on GE supplied local instrument racks were installed during the Unit 1 preoperational testing phase prior to receipt of the Low Power Operating License with the approval of General Electric, the Nuclear Steam Supply System (NSSS) supplier, under Field Deviation Disposition Request (FDDR) HH1-4167. The FDDR approved valve installation but left material qualification and procurement of proper materials (i.e. valves) as the Philadelphia Electric Company's responsibility. This responsibility was not recognized. In addition, the valves installed on non-GE supplied local instrument racks were also installed prior to receipt of the Low Power Operating License using the guidance supplied by the FDDR, but without formal design approval by the A/E due to a similar error on the part of the plant staff supervision.

In summary, the installation of these valves failed to comply with either the A/E or the PECo modification procedures in effect at the time. This is due to the multiple modification procedures in use prior to receipt of the low power license and the failure of a plant staff engineer to ensure proper material selection for the modification. A detailed root cause analysis will be performed by August 31, 1989, to determine whether further investigation into the generic concern is necessary.

Corrective Actions:

All nineteen (19) test valves required to be removed were removed by Instrumentation and Control technicians under Maintenance Request Forms by May 13, 1989. In response to the disposition to the Nonconformance Reports the additional ninety (90) valves left in place will be administratively controlled as follows:

1) A routine test, RT-2-000-631-1, was written to perform required leak inspections of the remaining ninety (90) NUPRO instrument test valves until the third Unit 1 refuel outage.

2) The number of valve cycles is being tracked through Instrument and Control Department aid tags hung on each remaining NUPRO test valve.

3) Plant staff is initiating Modification requests to remove and replace the remaining valves prior to restart from the Unit 1 third refueling outage.

Actions Taken to Prevent Recurrence:

Upon receipt of the Unit 1 Low Power Operating License, Administrative Procedure A-14 "Procedure for Control of Plant Modifications," was placed in effect and adequately provided instruction and control throughout the modification process. This procedure addresses the modification review process and involves the independent review of several specialized work groups, supervision and management. Plant staff has determined that the current modification process is adequate and provides the proper instruction to attain the appropriate independent reviews. A detailed root cause analysis will be performed by August 31, 1989, to determine whether further investigation into the generic concern is necessary.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Limerick Generating Station, Unit 1	0 5 0 0 0 3 5 2	8 9	— 0 3 4	— 0 1	0 8 OF 0 8

TEXT (If more space is required, use additional NRC Form 366A's) (17)

Previous Similar Occurrences:

None

Cause Code: D2 Inadequate Procedure
A2 Failure to follow implementing procedures