GPU NUCLEAR

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January 31, 1997
6730-97-2043

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

Dear Sir:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
Licensee Event Report 97-01; Seven Drywell Penetrations Do Not Meet the Requirements described in Generic Letter 96-06

Enclosed is Licensee Event Report 97-01. This event did not impact the health and safety of the public.

If any additional information or assistance is required, please contact Mr. John Rogers of my staff at 609.971.4893.

Very truly yours,

Michael B Roche

Michael B. Roche
Vice President and Director
Oyster Creek



MBR/JJR
Enclosure

cc: Oyster Creek NRC Project Manager
Administrator, Region I
Senior Resident Inspector

IE22/1

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

# LICENSEE EVENT REPORT (LER)

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98



| FACILITY NAME (1) | DOCKET NUMBER (2) | PAGE (3) |
|---|---|---|
| Oyster Creek Unit 1 | 05000 - 219 | 1 of 4 |

TITLE (4)

Seven Drywell Penetrations Do Not Meet the Requirements Described in Generic Letter 96-06

| 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 |
| | | | | | | | | | | 05000 |
| 01 | 03 | 97 | 97 -- | 01 -- | 00 | 01 | 31 | 97 | FACILITY NAME | DOCKET NUMBER 05000 |

| | | THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11) | | | |
|---|---|---|---|---|---|
| OPERATING MODE (9) | N | 20.2201(b) | 20.2203(a)(2)(v) | 50.73(a)(2)(i) | 50.73(a)(2)(viii) |
| POWER LEVEL (10) | 100 | 20.2203(a)(1) | 20.2203(a)(3)(i) | X 50.73(a)(2)(ii) | 50.73(a)(2)(x) |
| | | 20.2203(a)(2)(i) | 20.2203(a)(3)(ii) | 50.73(a)(2)(iii) | 73.71 |
| | | 20.2203(a)(2)(ii) | 20.2203(a)(4) | 50.73(a)(2)(iv) | OTHER |
| | | 20.2203(a)(2)(iii) | 50.36(c)(1) | 50.73(a)(2)(v) | |
| | | 20.2203(a)(2)(iv) | 50.36(c)(2) | 50.73(a)(2)(vii) | |

LICENSEE CONTACT FOR THIS LER (12)

| NAME | TELEPHONE NUMBER (Include Area Code) |
|---|---|
| Peter Tamburro | 609.971.4141 |

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 | MONTH | DAY | YEAR |
|---|---|---|---|---|---|
| YES (If yes, complete EXPECTED SUBMISSION DATE). | X NO | | | | |

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On January 3, 1997, during a review as requested by Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions", seven penetrations in five systems did not meet the requirements as described in GL 96-06. Under the postulated conditions contained in GL 96-06, these seven penetrations are only required for containment integrity. An analysis was performed of each penetration while isolated. The analysis modeled the effects on internal fluid and piping in response to an external ambient temperature increase. The results revealed that although the piping did not meet the design requirements, the postulated pressures did not exceed ASME Section III, Appendix F criteria for piping. Additionally, the potential effects on the respective isolation valves were considered. A catastrophic failure of the valves is not considered credible. Therefore containment integrity for the penetration was maintained and the safety significance of this discovery is considered minimal. The cause for this condition was that previous analyses were performed without using the more conservative assumptions described in GL 96-06. Operability determinations were performed and further evaluations are ongoing to determine the need for modifications or procedural revisions.

| FACILITY NAME (1) | DOCKET (2) | LER NUMBER (6) | | | PAGE (3) |
|---|---|---|---|---|---|
| | | YEAR | SEQUENTIAL NUMBER | REV | |
| Oyster Creek, Unit 1 | 05000 -219 | 97 | 01 | 00 | 2 of 4 |

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

## DATE OF DISCOVERY

The conditions described in this report were discovered on January 3, 1997.

## IDENTIFICATION OF DISCOVERY

During a review requested by Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions", seven penetrations (EIIC - PEN) in five systems were evaluated as not meeting the requirements presented in GL 96-06. The systems are Shutdown Cooling (SDC)(EIIS - KE), Reactor Building Closed Cooling Water (RBCCW)(EIIS - KG), Reactor Water Cleanup (RWCU)(EIIS - KH), Isolation Condenser (IC)(EIIS - BL), and Reactor Recirculation Loop Sampling (RLS)(EIIS - KN). This discovery is considered reportable under 10 CFR 50.73(a)(2)(ii).

## CONDITIONS PRIOR TO DISCOVERY

The reactor was operating at approximately 100% power. At the time of discovery, system pressures and temperatures were normal for full power operation. However, the plant had been operated in all modes with these conditions since original plant startup.

## DESCRIPTION OF DISCOVERY

GL 96-06 requested licensees to consider the possibility of equipment damage following a design basis accident caused by the heating of trapped internal fluids. During the review of affected plant systems, it was determined that seven penetrations did not meet the design stress requirements. The areas of concern are: 1) the supply and return penetrations for the SDC system; and 2) the RBCCW return line penetration. Additionally, the following penetrations become concerns if they were previously isolated for maintenance: 1) The RWCU supply line penetration; 2) both IC return line penetrations; and 3) the RLS penetration.

## DESCRIPTION OF DISCOVERY (Cont.)

The review of the isolated piping sections revealed that following a design basis Loss of Coolant Accident (LOCA) or Main Steamline Break accident (MSLB), heating of the fluid trapped in the sections would cause thermal expansion. This expansion increases the internal pressure of the fluid in the piping, and results in the seven identified piping sections exceeding ASA/ANSI B31.1 Code allowable stresses.

## APPARENT CAUSE OF THE OCCURRENCE

The root cause of the condition is that previous analyses were performed without using the more conservative assumptions described in GL 96-06.

## ANALYSIS OF DISCOVERY AND SAFETY ASSESSMENT

During a review requested by Generic Letter (GL) 96-06, seven penetrations in five systems were evaluated as not meeting the requirements presented in GL 96-06. The five systems are not required to operate to mitigate LOCA or MSLB accidents. Therefore, the sole function of these seven penetrations is for containment integrity.

A review utilizing the guidance provided in GL 91-18,"Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability", was conducted to evaluate the operability of these seven penetrations. The review assumed that each penetration was isolated. The analysis modeled the effects on internal fluid and piping in response to an external ambient temperature increase. Actual configuration and material properties were utilized. The results revealed that the piping did not meet the design requirements. However, even when the most conservative assumptions were included, the postulated pressures did not exceed ASME Section III, Appendix F criteria for piping. Therefore containment integrity for the piping was maintained.

An analysis of the isolation valves (EIIC - IV) was then performed. The SDC, RWCU, IC, and RLS valves are all either 600 psi class or 900 psi class valves. No credit was taken for seat leakage. However, it was determined that valve bonnet or packing leakage was likely. This would relieve the pressure in the penetration. Based on a review of the design, a catastrophic failure of any of these valves is not considered credible.

## ANALYSIS OF DISCOVERY AND SAFETY ASSESSMENT (Cont.)

The RBCCW isolation valves were evaluated. These are six inch 150 psi class gate valves with bolted bonnets and solid disks. Discussion and correspondence with the valve vendor indicated that the valves leak past their seats when overpressurized on the upstream side. Significant overpressurization upstream of the valve causes pressurization of the valve bonnet, which results in a curvature of the flat body neck. Resultingly, this forces the body seat out of plane and allows the valve to leak. Once the leakage has reduced pressure, the forces on the valve neck reduce and the valve would reseat. Therefore, catastrophic failure of these valves in not considered credible.

Based on the fact that: 1) these five systems are not required to operate to mitigate a LOCA or MSLB; 2) the penetration pressures do not exceed ASME Section III, Appendix F criteria; and 3) the affected valves will not fail catastrophically, the safety significance of these concerns are minimal.

## CORRECTIVE ACTIONS

**Immediate Actions Taken**
Upon discovery of this concern, an operability determination was performed which determined that the isolated penetrations would maintain containment integrity.

**Short Term Compensatory Actions**
Two SDC valves outside the containment were opened to reduce the peak pressure of the penetration during the LOCA or MSLB.

**Long Term Corrective Actions**
The seven penetrations will be further evaluated to determine if modifications or procedural revisions are required.

## SIMILAR EVENTS

LER 96-015; Reactor Water Cleanup Valves May Not Operate During a Line Break Due to a Non Conservative Analysis

LER 95-005; Non Conservative Anticipatory Scram Bypass Switch Deficiency due to Original Plant Design