NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

# LICENSEE EVENT REPORT (LER)

| FACILITY NAME (1) | DOCKET NUMBER (2) | PAGE (3) |
|---|---|---|
| Oyster Creek, Unit 1 | 0 5 0 0 0 2 1 9 | 1 OF 0 7 |

TITLE (4)
Emergency Service Water-Containment Spray Negative Delta Pressure

| EVENT DATE (5) MONTH | DAY | YEAR | LER NUMBER (6) YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | REPORT DATE (7) MONTH | DAY | YEAR | OTHER FACILITIES INVOLVED (8) FACILITY NAMES | DOCKET NUMBER(S) |
|---|---|---|---|---|---|---|---|---|---|---|
| 1 1 | 1 0 | 7 7 | 8 4 | 0 2 6 | 0 0 | 1 1 | 2 0 | 8 4 | -- | 0 5 0 0 0 |
| | | | | | | | | | | 0 5 0 0 0 |

OPERATING MODE (9): N
POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (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) | 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) | 50.73(a)(2)(i) | 50.73(a)(2)(viii)(A) | |
| 20.405(a)(1)(iv) | X 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 | |
|---|---|---|
| Paul F. Czaya, Licensing Engineer | 6 0 9 | 9 7 1 - 4 8 9 3 |

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 |
|---|---|---|---|---|---|
| X YES (If yes, complete EXPECTED SUBMISSION DATE) | NO | | 0 6 | 3 0 | 8 5 |

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

Surveillance test results since November 10, 1977 indicate that the delta pressure between the Emergency Service Water (tube side) and Containment Spray (shell side) is negative tube-to-shell in the Containment Spray heat exchangers. This condition is contrary to that described in the Facility Description and Safety Analysis Report. A negative tube-to-shell delta pressure would allow leakage from the Containment Spray (CS) System to enter the Emergency Service Water (ESW) System at the CS heat exchanger and, thence, to the environment. The cause of the negative delta pressure is believed to be a decrease in ESW pump performance and increased pressure drop in ESW piping. Radiological consequences during normal operation and accident conditions have been analyzed and the contribution to offsite dose is considered negligible. Post-LOCA containment heat removal characteristics of the ESW/CS System are not affected. Corrective action will be determined pending the outcome of ongoing ESW System evaluation.



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## DATE OF OCCURRENCE

The earliest record identifying this condition is a surveillance test data sheet dated November 10, 1977.

## IDENTIFICATION OF OCCURRENCE

The Emergency Service Water (ESW) side of the Containment Spray (CS) heat exchangers operates at a lower pressure than the CS side.

This condition is reportable in accordance with 10CFR50.73 (a)(2)(ii)(B).

## CONDITIONS PRIOR TO OCCURRENCE

The reactor was in various operating and shutdown modes prior to and subsequent to the identification of this condition.

## DESCRIPTION OF OCCURRENCE

The ESW System delivers cooling water to the tube side of the CS heat exchangers from the ultimate heat sink (Barnegat Bay via intake canal). The CS System circulates demineralized water from the pressure suppression chamber (Torus) through the shell side of the CS heat exchangers to the Drywell for post-LOCA containment cooling. Installed delta pressure instrumentation provides indication of tube-to-shell side delta pressure in the CS heat exchangers.

Since November 10, 1977, the tube-to-shell delta pressure has been recorded as less than zero on surveillance test data sheets. This condition is contrary to the system design basis as stated in Section VI-7.2 of the Facility Description and Safety Analysis Report (FDSAR) which states: "Each of the four service water pumps will also have a capacity of about 3000 gpm but will deliver a higher head than the containment spray pump to maintain a higher pressure on the service water side of the heat exchangers so that any leakage will be into the containment side of the heat exchangers." Data resulting from pre-operational testing shows that positive tube-to-shell delta pressure was achieved as designed.

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

## APPARENT CAUSE OF OCCURRENCE

The apparent cause of this condition is believed to be gradual degradation of ESW System performance since initial plant operation. Some biofouling found in ESW piping resulting in increased pressure drop in ESW piping coupled with a decrease in ESW pump performance characteristics result in decreased ESW pressure at the CS heat exchangers.

## ANALYSIS OF OCCURRENCE AND SAFETY ASSESSMENT

The function of the CS and ESW Systems is to remove heat from the primary containment following a LOCA to control pressure and temperature and maintain them at acceptable levels in order to: 1) reduce the driving force for containment leakage; 2) maintain the structural integrity of containment. The CS/ESW System transfers post-accident decay heat to the ultimate heat sink and provides a physical boundary to the release of post-accident fission products. The CS/ESW System comprises two redundant loops each containing two heat exchangers, two ESW pumps and two CS pumps. The safety function is achieved with operation of one CS pump, one ESW pump and two heat exchangers piped in parallel in one of the two loops.

The consequence of higher CS (shell side) pressure than ESW (tube side) pressure is that should a leak develop in the heat exchanger then Torus water would leak into the ESW System. Radioactive material contained in Torus water would then be discharged to the discharge canal through the leak.

The capability of the CS/ESW System to perform its cooling safety function is not affected by the existing system condition. Adequate cooling water flow is provided by ESW pumps. As determined by calculation the minimum required flow is equal to 2370 gpm. Surveillance testing continually demonstrates ESW flow to be approximately 3000 gpm or greater. Should heat exchanger leakage occur, it will be minimal and will not affect CS cooling effectiveness.

The capability of the CS/ESW System to provide a boundary to the release of post-accident fission products has been affected by the negative tube-to-shell CS heat exchanger delta pressure. Leakage in the heat exchangers will not be contained within the CS System. Leakage can escape the system boundary into the ESW System and, ultimately, to the discharge canal.

An evaluation has been performed to estimate the offsite dose due to leakage from the CS System during a loss of coolant accident (LOCA). The scenario for this evaluation considers a LOCA has occurred, primary coolant has been introduced into the Torus, and the CS System is initiated. Further, there is a leak in the CS heat exchanger and the pressure in the CS System is higher than in the ESW System thus introducing radioactivity to the environment.

Several assumptions were made and parameters chosen in evaluating this hypothetical leakage and the subsequent offsite dose. The source term utilized was derived from the core inventory given in the Wash-1400 Reactor Safety Study, Appendix 6, Calculation of Reactor Accident Consequences. The figure utilized was $3.98E^8$ curies. This activity was assumed to be all Iodine-131, even though the particulate activity was factored into the source term. Due to the release being unpressurized liquid, no consideration was given to noble gas activity, nor was consideration given to the various phases of iodine activity and their associated characteristics. All of the $3.98E^8$ curies was assumed to enter into the torus and be diluted by Torus water (82,000 cubic feet). This is considered conservative in that all of the activity would not enter into the Torus and the Torus water volume assumed is the minimum level allowed during plant operation. A further measure of conservatism was included in that no credit was taken for water in the CS System piping and heat exchangers. It was assumed that the CS System was in service for one week after the LOCA. The information is further detailed on Table 1 below:

Table 1

| | |
|---|---|
| Activity Released to Torus (Wash-1400) | $3.98E^8$ Curies |
| Torus Water Level | 82,000 cubic feet<br>$2.32E^9$ ML |
| Activity of Torus Water | $1.72\ E^5$ uCi/ML |
| Leak Rate | 0.03 gallons/hour<br>$1.9E^4$ ML/week |
| Total Activity Released | $3.27E^3$ Curies |
| Dilution (Discharge Canal Flow) | 2138.9 cubic feet/second |

The information in Table 1 was entered into a dose analysis computer program. This program computes dose for several pathways. The three pathways utilized for the dose calculation were: 1) exposure from shoreline deposition; 2) ingestion of salt water sport fish and; 3) ingestion of salt water invertebrates. These are the most likely pathways of offsite exposure from a spill into the discharge canal. Dose due to each exposure pathway was calculated and then summed for each age group. The program is in accordance with Regulatory Guide 1.109.

As the dose model used considered exposure factors appropriate for evaluations during normal plant operations (per Regulatory Guide 1.109) potential radiation dose values are, therefore, conservative. There was no credit taken for actions which would be performed to limit exposure from the pathways identified. Plant procedures are in effect that specify actions to be taken to minimize offsite dose during an occurrence such as this hypothetical leak. Emergency Plan Implementing Procedure (EPIP) No. 1 will be implemented upon receipt of a high radiation alarm at the discharge of the heat exchanger (ESW side). Since the radiation monitors are area radiation monitoring devices, their indications may not be entirely reliable. However, sampling will also be performed in the discharge canal in accordance with EPIP-11 to determine appropriate actions to minimize exposure to radioactive effluents. These actions would include recommendations to restrict fish and invertebrate ingestion if such protective actions are warranted and the isolation of the leaking containment spray heat exchanger based on ESW radioactivity levels as determined via sampling. Limits of the sampling are based on 10CFR20 limits, which are more conservative than 10CFR100 limits. Notification of civil authorities and the activation of the Environmental Assessment Command Center will be implemented by EPIP-3 and EPIP-35. Protective action recommendations will be made per EPIP-2 to appropriate civil authorities to ensure that actions such as fish/invertibrate ingestion are implemented to ensure that any effect to the health and safety of the public is minimized. These controls are enacted to minimize the ingestion pathway and will occur if radioactivity is detected in the canal. Therefore, including ingestion pathway doses is inappropriate for this evaluation. This is consistent with Regulatory Guide 1.3, which does not include the ingestion pathway doses post-LOCA.

Integral to the evaluation of the acceptability of the present system condition is the integrity of the tube and shell sides of the CS heat exchangers. The original heat exchangers were replaced with ones containing 90-10 copper-nickel alloy tubes during the 1978-79 refueling outage. These tubes developed pinhole leaks in approximately one year and were replaced with titanium tubes in 1980. The heat exchanger presently is constructed of titanium tubes with an aluminum-bronze tube sheet. The corrosion resistance of these materials against sea water is excellent. Therefore, the probability of any gross failure of tubing is small. To ensure against an existing leak, the heat exchangers have been leak tested before restart from the recent refueling outage. The criteria for allowable leakage during the test was based on a small fraction of the exposure limits of 10CFR100 as outlined below. The dose evaluation model used in calculating doses is based on normal operations when ingestion of fish and invertibrates is not restricted. However, since dose due to ingestion will not occur during an accident because of the plant controls discussed previously, the only dose factor will be direct shoreline exposure. As noted in Table 1, the leak rate model used a 0.03 gph CS water leak rate into ESW as being representative of a leak due to galvanic corrosion between the tube and tubesheet from all four heat exchangers. To be more conservative and to specify a leakage criterion that can be accurately measured, a leak rate of 1 gpm was used as the input to the dose calculation. This increases doses by a factor of approximately 2000. Using the 1 gpm leakage and recalculating direct exposure dose the following table results:

Table 2

Offsite Doses Based on CS System Leakage

Scaled to a Leak Rate of 1 gpm

| | Direct Shoreline Exposure (Rem) | 10CFR100 Limits (Rem) |
|---|---|---|
| Whole Body | 0.024 | 25 |
| Thyroid | 0.024 | 300 |

As stated in "Safety Evaluation of SEP Topic XV-19, Radiological Consequences of a Loss of Coolant Accident", the potential contribution to the LOCA dose of less than 0.1 rem is considered to be negligible. Therefore, it is concluded that the doses caused by CS heat exchanger leakage is negligible. Thus, 1 gpm was the criterion for leakage permitted during the CS heat exchanger leak test although a higher limit could be justified through this same logic. This leak rate limit applies to the total of all four heat exchangers and still provides a factor of 4 margin to what has previously been identified as an insignificant dose. The CS System is the only system operating after a LOCA which has this potential for liquid release to the environment.

The existing condition will not create significant environmental impact during normal operations. The CS heat exchanger is in operation only for surveillance testing during normal operations. Surveillance testing of both the CS and ESW systems is performed once per month. During the surveillance test, both CS and ESW systems operate simultaneously with one pump operating in each system. One loop is tested at a time. The pumps run for about 15 to 20 minutes, during which time they stabilize and the readings for flow rates and pressures are taken. If a CS heat exchanger develops a leak during testing, leakage to the environment could occur. However, since the duration of the test is very short and the Torus water chemistry and radiation level is on the order of $10^{-4}$ uCi/ml (liquid isotopic concentrations) the contribution from leakage to offsite releases will be approximately nine orders of magnitude less than the values calculated for the accident scenario, and therefore will be negligible.

## CORRECTIVE ACTION

During the recent refueling outage steps were taken to assess the operational condition of the ESW System as follows:

1) An examination of selected portions of ESW piping was performed. Some biofouling was noted near the ESW pumps at the intake structure, but the extent of the biofouling was considered to not have a significant effect on flow rate. ESW piping examined in the turbine and reactor buildings was found to be clean. ESW piping is chlorinated on an intermitent basis to minimize further biofouling.

2) One of the four ESW pumps was refurbished. A minor improvement in performance was noted, however, indicated performance characteristics are still below the design pump curve.

3) As it is felt that current ESW flow instrumentation might provide a conservative indication of actual ESW flow rate, an annubar flow measuring device was installed in one ESW loop to verify the accuracy of portable ultrasonic flow instrumentation. However, problems were encountered in obtaining a reliable measurement with the annubar. Efforts will continue to resolve these hardware problems.

4) Two of four restricting orifices in ESW discharge lines from the CS heat exchangers were examined and found to be in good condition. Their correct sizing was verified.

Further corrective action will be based upon the ongoing evaluation of ESW System performance.

The CS heat exchangers were checked for leaks prior to entering the plant's post-outage startup phase and were found to have no leakage.

GPU Nuclear

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November 20, 1984

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

Dear Sir:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
Licensee Event Report

This letter forwards one (1) copy of Licensee Event Report (LER) No. 84-026. It is not known how long the condition described herein has existed. Surveillance records date back to November 1977 and the condition was identified in IE Inspection Report No. 78-19. Previously, it was not believed to be reportable. In view of current LER reporting criteria, we consider it appropriate to report the condition at this time.

Very truly yours,

Peter B. Fiedler
Vice President and Director
Oyster Creek

PBF/PFC/dam
Enclosures

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