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January 20, 1997
6730-97-2032

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

Enclosed is Licensee Event Report 96-015. 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

I/1
IE22

MBR/JJR
Enclosure

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

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

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

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), 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)

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TITLE (4)

Reactor Water Cleanup Isolation Valves May Not Operate During A Line Break Due to Non-Conservative Analysis

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
Month	Day	Year	Year	Sequential Number	Revision	Month	Day	Year	Facility Name	Docket Number
12	20	96	96 --	015 --	00	01	20	97	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)			
POWER LEVEL (10)	100	20.2201(b)	20.2203(a)(2)(v)	50.73(a)(2)(i)	50.73(a)(2)(viii)
		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)	Specify in Abstract below or in NRC Form 366A
		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)
Thomas S. Corcoran	201-316-7262

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 December 20, 1996, during the review of a vendor service letter, it was identified that the thermal overloads for the starters for the Reactor Water Clean Up (RWCU) Isolation Valves were not sized for a revised Reactor Building environment following a High Energy Line Break from the RWCU System. This condition is considered reportable under 10 CFR 50.73(a)(2)(ii). The root cause of this deficiency is that previous HELB analyses were performed without using the most conservative assumptions.

The safety significance of this condition has been determined to be minimal. The reactor is normally operated at 100% power. Above approximately 88% power, a reactor scram is received with any break size larger than approximately 50% of the cross sectional area of the pipe. Below 50% break size, the analysis is bounded (both temperature profile and offsite release) by existing analyses. Immediate corrective actions were taken to provide the operators with guidance on addressing the potential line break. The thermal overloads were bypassed to ensure the ability to isolate the system. A reanalysis for other potentially affected Safety Related equipment was commenced to ensure continued operability for required systems. Modifications to provide an automatic RWCU isolation on a line break will be evaluated for possible installation.

DATE OF DISCOVERY

The condition described in this report was identified on December 20, 1996.

IDENTIFICATION OF OCCURRENCE

During a review of General Electric Service Information Letter (SIL) 604, Reactor Water Clean Up System (RWCU) Break Detection for applicability to Oyster Creek, it was identified that the thermal overloads for the starters for RWCU Isolation Valves V-16-1 and V-16-61 were not sized for a revised Reactor Building environment following a High Energy Line Break (HELB) from the RWCU System under certain conditions.

This condition is considered reportable under 10 CFR 50.73(a)(2)(ii).

CONDITIONS PRIOR TO DISCOVERY

The plant 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 this condition since initial startup.

DESCRIPTION OF OCCURRENCE

General Electric Service Information Letter (SIL) 604 identified a concern that Reactor Water Clean-Up (RWCU) System pipe break analysis may be non-conservative. Under certain conditions, the system isolation logic might not isolate the postulated break. The licensing and design bases for the original radiological release and Environmental Qualification analyses for pipe breaks outside containment assumed full power operation and a double-ended guillotine break of the largest pipe with feedwater available.

With the feedwater system in operation at greater than 88% power, a RWCU line break causes reactor water level to reach the Low Reactor Water Level scram setpoint thus shutting down the reactor. As a result of the scram, the reactor depressurizes through the break producing a decrease in break flow rate. Under the postulated SIL conditions of reduced power and/or reduced break area, the reactor level may not reach the low level scram setpoint.

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TEXT CONTINUATION

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

DESCRIPTION OF OCCURRENCE (Cont'd)

Analysis showed that at reduced reactor power or reduced break size, the feedwater system is capable of maintaining the reactor water level above the low level setpoint, thus preventing the low level reactor scram. A computer analysis was used to assess the impact on the Reactor Building environment as well as the amount of reactor coolant mass released to the environment from the building. This evaluation showed that although the mass release in the building exceeds that associated with the previous RWCU line break analysis, it is still bounded by the existing Main Steam Line break documented in the Oyster Creek Final Safety Analysis Report (FSAR). Therefore, the offsite release is bounded by the Main Steam Line break calculation. However, the environmental conditions in the building are more severe than those previously calculated.

In design bases analyses, no operator action can be assumed for ten minutes. The revised secondary containment temperature profile was subsequently evaluated and it was determined that the thermal overloads for RWCU Isolation Valves V-16-1 and V-16-61 were not sized for the new projected temperature profile. This could cause the thermal overloads to trip prematurely, thus rendering the valves inoperable. The operator may be unable to close these valves after a ten minute time period to terminate the release.

APPARENT CAUSE OF OCCURRENCE

The root cause of this deficiency is that previous HELB analyses were performed without using the most conservative assumptions.

ANALYSIS OF OCCURRENCE AND SAFETY ASSESSMENT

The Reactor Protection System (RPS) (EIIS JC) monitors plant parameters and automatically initiates protective actions if established limits are exceeded. A prior analysis of RWCU line break at full power with feedwater available took credit for a decreasing level in the reactor to cause a low level scram, thereby decreasing system pressure and break flow rate. The resulting mass release was bounded by previous analysis. The secondary containment temperature profile was evaluated and found to be satisfactory.

The RWCU line break at reduced power or for less than a double-ended guillotine break requires operator action to mitigate the event. The operator would be required to isolate the break by closing system isolation valves V-16-1 and V-16-61. Under the newly analyzed condition, a RWCU line break could cause a temperature profile in secondary containment that could trip the thermal overloads on the RWCU isolation valve starters and the operator might be unable to isolate the leak.

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

ANALYSIS OF OCCURRENCE AND SAFETY ASSESSMENT (Cont'd.)

The safety significance of this condition has been determined to be minimal. The reactor is normally operated at 100% power. Analysis has shown that at above approximately 88% power, a reactor scram will occur with any break size larger than approximately 50% of the pipe cross sectional area. Below 50% break size, the analysis is bounded (both temperature profile and offsite release) by prior analyses. In addition, the Emergency Operating Procedures provide guidance on HELB response and the operators receive simulator training on RWCU line breaks inside secondary containment. The offsite release by the revised RWCU line break is bounded by the steam line break documented in the FSAR.

CORRECTIVE ACTIONS

Immediate Corrective Actions
Upon determination of the condition, interim guidance was issued to licensed operators to remove the RWCU System from service if power is reduced below 90%. Additionally, guidance was provided to the operators to isolate the RWCU System if the area temperature sensors near the RWCU System pipes alarmed.

Short Term Corrective Actions
Shortly thereafter, a plant modification was completed to bypass the thermal overloads on V-16-1 and V-16-61. Upon completion of the modification, the interim guidance on RWCU operation was removed.

Long Term Corrective Action
A full reanalysis of the Environmental Qualification for potentially affected Safety Related Components was initiated. To ensure continued operability for these components, the interim guidance was restored until the reanalysis is completed. Modifications to provide an automatic RWCU system isolation on a line break will be evaluated for possible installation.

FAILURE DATA

None

SIMILAR EVENTS

LER 95-005: Non Conservative Anticipatory Scram Bypass Switch Setpoint due to Original Plant Design