AmerGen

A PECO Energy/British Energy Company

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2130-00-20316
December 14, 2000

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

Dear Sir:

Subject: Oyster Creek Generating Station
Docket No. 50-219
License Event Report 2000-011
Reactor Scram Due To Low Reactor Water Level Resulting From Personnel Error

Enclosed is License Event Report 2000-011. This event did not effect the health and safety of the public and is not considered a Safety System Functional Failure for purposes of NRC performance indicator reporting. This LER contains no long term continuing commitments.

If additional information is required, please contact Brenda DeMerchant Oyster Creek Licensing Engineer, at 609.971.4642.

Very truly yours,

Ron J, DeGregorio
Vice President, Oyster Creek

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

IE22

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98
LICENSEE EVENT REPORT (LER)	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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Oyster Creek Unit 1	05000 - 219	1 of 5

TITLE (4)

Reactor Scram Due To Low Reactor Water Level Resulting From Personnel Error

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
11	15	2000	20 00	-- 11	-- 00				FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	R
POWER LEVEL (10)	5%

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

	20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(I)		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)	X	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)
Brenda DeMerchant	609.971.4642

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 November 15, 2000, during reactor startup from the 18 R outage, turbine pre-warming had been completed and Operators were transitioning to steam chest warming in accordance with procedures. Reactor Pressure was approximately 460 psig. Due to a misinterpretation of the procedure, the operators did not close the Bypass Valve Opening Jack prior to closing the Load Limiter. This resulted in several Turbine Bypass Valves (BPVs) rapidly opening and a subsequent reactor level increase. Immediate response to the high level condition was to reduce feedwater flow and increase letdown. This action combined with the inventory loss through the BPVs resulted in overcompensating and the level decreased to the low-level scram setpoint. A full reactor scram was received. All control rods inserted. Reactor level was stabilized. Reactor pressure continued to decrease until the bypass valves were shut (approximately 17 minutes into the transient)

The main consequence of this transient was the scram, which resulted in the excessive cooldown rate, and exceeded the 100 degrees F per hour Technical Specification limit.

The root cause of this event was personnel error. Operators demonstrated a lack of fundamental knowledge with respect to system operations in that they did not fully comprehend the consequences of not closing the bypass valve opening jack prior to closing the limit load and failed to close the BPVs.

Corrective actions included revising the procedure to clarify the misinterpreted step. An evaluation of the effects of excessive cool down rate was also conducted.

DATE OF DISCOVERY

This event occurred on November 15, 2000 at 1748 hours.

IDENTIFICATION OF OCCURRENCE

During startup from the 18R outage, while prewarming of the main Turbine (EIIS-TA) was in progress, the load limit was closed prior to closing the bypass valve opening jack. This caused a drop in reactor pressure and subsequent level swell. During level restoration actions a low-level scram occurred. All control rods inserted. Reactor level was stabilized. Reactor pressure continued to decrease until the bypass valves were shut.

This event is considered reportable in accordance with 10 CFR 50.73 (a)(2)(iv) as well as 10 CFR 50.73 (a)(2)(i)(B)

CONDITIONS PRIOR TO DISCOVERY

A plant startup was in progress with reactor power high in intermediate range monitor (IRM) 8/ low in IRM range 9. Reactor pressure was approximately 460 psig. Plant heatup was in progress.

DESCRIPTION OF OCCURRENCE

On November 15, 2000, the reactor was critical in IRM range 8 and a plant heatup was in progress. The plant was starting up from the 18R outage. Operators were directed to transition to steam chest warming in accordance with procedures. At this point operators closed the #2 main stop valve internal bypass. The next step in the procedure is written as an If/Then statement, which was interpreted by the operators as being not applicable (the bypass valve opening jack was not controlling pressure). Based upon this interpretation the operators did not close the bypass valve opening jack and proceeded to close the load limit, closing the control valves. In response to the control valves going closed, the bypass valves opened (approximately 1-1/2 to 2 valves opened). This resulted in a reactor water level swell. As reactor pressure decreased a reactor Hi level alarm was received. The operators responded to the high reactor level alarm by reducing feed flow and increasing reactor water cleanup letdown flow but did not close the BPV's. These actions were taken to prevent a Turbine trip on reactor high level. As reactor level turned the Operators secured reactor water cleanup letdown, (Clean-up system tripped on system high pressure) and attempted to increase reactor feed flow to the vessel using the low flow valves. Reactor level continued down due to inventory loss via the BPVs. The reactor low level scram set point was reached and a reactor scram was received. All control rods inserted. Reactor level was stabilized. Reactor pressure continued to decrease until the bypass valves were shut (approximately 17 minutes into the transient).

The main consequence of this transient was the scram, which resulted in the excessive cooldown rate, and exceeded the 100 degrees F per hour Technical Specification limit.

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

Apparent Cause of Occurrence

This event was caused by personnel error. Based on operator interviews and review of the transient data it appears that operator actions were less than adequate. A loss of oversight and command and control resulted in the shift focusing on the reactor level transient and not properly controlling reactor pressure. The Site Shift Manager allowed himself to become overly involved with the task of turbine warming. This distracted him from his control room oversight function. With the Site Shift Manager actively involved in the startup activities, command and control responsibilities were not clearly delineated.

Additionally a lack of fundamental knowledge with respect to system operation was identified in that operators did not fully comprehend the consequences of not performing the procedural step of closing the bypass valve opening jack prior to closing the Load Limit. A lack of fundamental knowledge with respect to integrated system response was also noted. After the bypass valves opened and reactor pressure decreased causing a reactor level transient, operators were focused on recovering reactor level as opposed to controlling reactor pressure.

A contributing cause to this event was determined to be procedural inadequacy. The step in question was written in an If/Then format and did not direct the operators to close the bypass valve opening jack prior to closing the load limit. The step should have directed the operators to confirm the bypass opening jack was closed. A problem had been identified earlier with this portion of the turbine warming procedure, however this concern was not dispositioned in a timely manner. Additionally, the evolution of turbine warming was not covered during the 'Just In Time' training nor was the training given to the supervisors on shift.

These actions resulted in a reactor scram and excessive reactor cooldown.

ANALYSIS OF OCCURRENCE AND SAFETY ASSESSMENT

The main consequence of this transient was the scram, which resulted in the excessive cooldown rate that exceeded the 100 degrees F per hour Technical Specification limit.

A review of the transient determined that the hourly cooldown rate was 111 degrees F/hr.for several minutes, which exceeded Technical Specification Limiting Conditions for Operation. Stresses induced as a result of this transient are proportional to the differential temperatures within the reactor metal. Technical Specifications allow 10 cooldowns exceeding 300 degrees F/hr. A calculation was performed in 1998 to redefine the allowable number of heatup/ cooldown and other design cycles for the reactor vessel by recalculating the usage factors. This calculation documents that 5 emergency cooldowns of 300 degrees F/hr are allowed as of the date of the calculation. A comprehensive review of LERs back to the date of the calculation in 1998 found no occurrences of exceeding the 100-degree F/hr cooldown. Therefore, there have been no cooldowns which would have used a portion of the fatigue usage attributed to the 300 degrees F/hr cooldown used in the calculation.

ANALYSIS OF OCCURRENCE AND SAFETY ASSESSMENT (con't)

The reactor pressure-temperature (P-T) curves identified in the plant technical specifications are based on a 100 degree F/hr heatup or cooldown rate. Using the "B" P-T curve for 32 EFPY (being conservative), the minimum required reactor vessel metal temperature is 220 degrees F at 460-psig-reactor pressure. During this transient, the minimum recirculation piping temperature was 337 degrees F at 460 psig, because this transient was a depressurization event with the reactor pressure following the steam saturation curve. This margin in temperature is enough to compensate for exceeding the 100-degree F/hr heatup or cooldown rate.

As for the reactor vessel fatigue management program, this transient will be treated as an upset condition, not an emergency condition and will be added to the normal cycle tracking list. It will not count against the 5 emergency cooldowns of 300 degrees F/hr allowed by the 1998 calculation. However, it does count against the normal heatup and cooldown cycles (240 allowed). Therefore, the fatigue usage as a result of this cooldown is not a significant concern. Fatigue management documentation will be updated to account for this event in order to maintain an accurate history.

In conclusion, the transient can be viewed as one within the analyzed basis for vessel thermal cycles, and therefore the safety significance of this event is considered to be minimal.

Corrective Action

The following corrective actions were done immediately after the event:

Procedure 315.1 (Main Turbine Operation) was revised to clarify step 3.3.15 and a review of the entire procedure was conducted to identify any additional problems.

An evaluation of the effects of excessive reactor cool down rate was completed.

A pre-shift briefing describing the event and operator response to the transient was conducted with all operating shifts. This briefing included a review of the roles and responsibilities of shift personnel and management's expectations with regards to error free behaviors.

Corrective Actions (con't)

The Senior Operations Manager discussed the event and the operator's response to the transient with the supervisors directly involved in the event. The briefing included a review of the responsibilities of shift personnel and management's expectations with regards to error free behaviors, especially command and control.

Upon a more detailed and comprehensive review of the crew's performance using input from on-site an off-site resources, the operating crew was disqualified and placed in a remediation program.

Longer term corrective actions:

An evaluation of shift compliment/assignments based on performance and shift assignments will be completed by the end of the year.

A review of the event on the simulator with a focus on specific behaviors that did not meet management's expectations will be conducted with all operating crews.

Specific systems training on Turbine Controls and related systems will be conducted.

The Fatigue Management documentation will be updated to account for this event in order to maintain an accurate history.

A station procedure revision will be made requiring individual accountability for identifying and resolving procedural problems in a timely manner.

The above corrective actions will be completed by the end of the first quarter 2001.

Similar Events

LER 84-033: Reactor Scram on Low Condenser Vacuum

LER 92-010 Low Reactor Water Level Scram During Startup Caused by a Turbine Valve Adjustment Change Due to a Loose Locking Device

LER 96-005: Reactor Scram on Low Water Level Due to Personnel Error