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NUCLEAR

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August 6, 1997
6730-97-2208

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

Dear Sir:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
Inspection Report 97-03
Reply to a Notice of Violation

By letter dated July 3, 1997, the NRC docketed Inspection Report 50-219/97-03. Enclosure I to that Report contained two Notices of Violation. Attachment I to this cover provides the requisite replies to the two violations. Please note that Violation 2 has been contested. Additional information has been provided for your review.

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

Very truly yours,

for Michael B. Roche
Vice President and Director
Oyster Creek

IE01

MBR/JJR
Attachment

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

Attachment 1

<u>NRC Notice of Violation 1</u> (four parts)

Technical Specification 6.8.1 requires that written procedures shall be established, implemented and maintained that meet or exceed the requirements of NRC Regulatory Guide 1.33.

Contrary to the above, procedure requirements were not established, implemented and maintained, as indicated by the following examples.

<u>NRC Notice of Violation 1.a.</u>

NRC Regulatory Guide 1.33, Appendix A (4.e) recommends procedures for activities, including those for startup, operation, and shutdown of the shutdown cooling system.

Procedure 305, "Shutdown Cooling System Operation," step 4.3.18, requires that <u>WHEN</u> flow in the operating loop has stabilized, <u>THEN</u> valve in pump suction pressure interlock for the operating pump by opening the respective pressure switch isolation valve. Step 5.5.13 of procedure 305 similarly requires that <u>WHEN</u> flow in the loop has stabilized, <u>THEN</u> place the suction pressure interlock for the selected pump in service by opening the respective pressure switch isolation valve.

Contrary to the above, on April 23, 1997, the control room operators placed the A and B loops of the shutdown cooling system in service without placing the suction pressure interlocks in service for the A and B pumps by opening the respective pressure switch isolation valves. As a result, the A and B pumps were operated for about 9 1/2 hours without the necessary low pump suction pressure automatic protection.

<u>GPUN Reply To Notice Of Violation 1.a.</u>

GPUN concurs with the violation as written.

While cooling down the reactor, the shutdown cooling (SDC) system was placed in service. The sequence to place the SDC system in service was delayed due to the anticipation of exceeding the RBCCW pump suction temperature limit of 133 °F.

The procedure required the operator to verify adequate pump suction pressure for the SDC pump by unisolating the associated pump suction pressure switch (one per loop for three loops), then re-isolating the switch to prevent spurious trips while establishing system flow. The procedure then directed the operator to individually throttle system flow and close the minimum flow valves when flow reached 500 gpm. This was completed.

To continue in the SDC procedure at this time would have caused the Reactor Building Closed Cooling Water (RBCCW) system outlet temperature to exceed the 133°F limit. Therefore, the reactor cooldown rate was established at this time using the RBCCW outlet valve. At this point in the procedure, the SDC suction pressure switches were still isolated.

A shift change occurred. As the SDC system had not yet reached its design flow rate of 3000 gpm, no significant changes in SDC operation or lineup were required, and the suction pressure switches remained isolated.

A second shift change occurred. Reactor temperature was approximately 160 °F with recirculation pumps C and D in operation. At this point, the isolated suction pressure switch tripped the B SDC pump. The actual suction pressure was above that required for pump operation, based on reactor water level.

Reason for the Violation

The root cause of the violation was an inadequate shift turnover. The status of the suction pressure switched (isolated) was not communicated to the oncoming crews. A contributory cause was identified in that an apparent conflict was found in the SDC and RBCCW operating procedures. Following the SDC procedure can result in operating the RBCCW system beyond its design temperature.

Corrective Steps Taken and Results Achieved

The operating shift quickly diagnosed the cause as the isolated suction pressure switch, unisolated the three switches, and restarted the B SDC pump.

Corrective Steps to be Taken to Prevent Recurrence

Shift briefings will be held to communicate the details of this event and emphasize the consequences of improper turnover of critical plant information. The SDC operating procedure will be revised to address RBCCW temperature limits and to ensure that the suction pressure switches are placed in service when flow is established. These actions will be completed by October 30, 1997.

Date When Full Compliance was Achieved

Full compliance was achieved when the suction pressure switches were unisolated and returned to service on April 23, 1997.

NRC Violation 1.b

NRC Regulatory Guide 1.33, Appendix A (1.c) recommends procedures for activities, including those for equipment control.

Procedure 108, "Equipment Control," step 9.1.1, requires the control room operator to determine the appropriate isolation boundaries for work activities using Attachment 108-6 (Guidelines for Isolation Boundaries). Item 2.4 of Attachment 108-6 states that the effects on the system if a valve is physically moved during maintenance work shall be considered and additional isolation boundaries shall be added to the outage as appropriate. Step 10.1 of procedure 108 requires the licensed operations supervisor to review the switching order for compatibility with license requirements and station operating conditions.

Contrary to the above, on March 11, 1997, the control room operator did not determine an appropriate isolation boundary for maintenance to torus spray valve V-21-18 in the containment spray system. The valve was physically moved several times during the maintenance without adequately considering the effects on the system or adding additional isolation boundaries to the outage. As a result, the pressure suppression function of the torus was degraded during the times that maintenance personnel had V-21-18 opened (total of about 10 minutes). Also, the licensed operations supervisor failed to identify that the switching order was not compatible with license requirements and station operating conditions.

GPUN Reply To Notice Of Violation 1.b.

GPUN concurs with the violation as written.

Preventive Maintenance on torus spray valve V-21-18 was conducted in accordance with an approved repetitive task. Prior to this event, this task had been scheduled to work during an outage. When this task was rescheduled from an outage to online maintenance, a safety review was performed. The safety review incorrectly stated that this rescheduling did not have the potential to adversely affect nuclear safety. The safety review relied solely on the isolation boundary to prevent degradation to the containment suppression function. The request for the isolation boundary and subsequent reviews did not identify the nuclear safety concern and did not establish the proper boundaries

Reason for the Violation

The root cause of this event was determined to be an inadequate safety review which was performed when the maintenance task was converted from an outage task to an online maintenance task. Neither the preparer nor the reviewer had sufficient operating knowledge to identify the potential containment concern.

Contributory causes were identified in that operations personnel did not immediately recognize the safety implications of cycling the identified valve. Finally, there were no posted precautions to warn operations or maintenance personnel about the safety implication of cycling V-21-18.

Corrective Steps Taken and Results Achieved

The degradation in containment was identified and promptly corrected. A pre-shift briefing was issued to alert licensed and non-licensed operators to the potential paths that could exist during testing and result in short cycling of the drywell suppression capability. Caution labels were posted at containment spray valves and controls to ensure that a short cycling path would not be created. Repetitive task documents which direct maintenance on containment spray systems were reviewed to prevent short cycling of the system and assure Technical Specification compliance for containment.

Corrective Steps to be Taken to Prevent Recurrence

A method for reviewing work activities and preparing switching orders which will allow adequate time to perform thorough reviews and establish appropriate controls is being developed. This method will be finalized during the third quarter of 1997.

Date When Full Compliance was Achieved

Full compliance was achieved on March 11, 1997, when V-21-18 was closed, restoring torus suppression capability.

NRC Violation 1.c

NRC Regulatory Guide 1.33, Appendix A (7.a) recommends procedures for activities, including those for the control of radioactivity (for limiting materials released to environment and limiting personnel exposure).

Procedure 320.1, "Demineralized Water Transfer System," Step 3.2.3, states "Portions of the Demineralized Water System are contaminated. Per Safety Evaluation SE-000523-011, Demineralized Water cannot be used outside any contained radiological control area (i.e.; yard, flush water overboard) unless specifically sampled just prior to use and found to have no detectable activity."

Contrary to the above, on April 14, 1997, the demineralized water transfer system was used to flush the service water system radiation monitor (discharge flowpath) without finding that the demineralized water system had no detectable activity, resulting in a small release of potentially contaminated demineralized water to the discharge canal.

GPUN Reply To Notice Of Violation 1.c

GPUN concurs with the violation as written.

Reason for the Violation

The root cause of the violation was equipment failure which occurred when an administratively locked closed valve leaked by. A contributing cause was personnel error in that samples of the Demineralized Water System were not obtained prior to commencing the evolution. Flow was initiated when a valve downstream of the leaking locked closed valve was opened, and stopped when the pressure in the line dissipated.

Corrective Steps Taken and Results Achieved

The downstream valve was tagged closed and subsequently locked. A sample of the water in the service water radiation monitor was taken and found to have less than minimum detectable activity. The leaking valve was replaced on July 14, 1997.

Corrective Actions to be Taken to Prevent Recurrence

The event critique will be included in the Industrial Events Training for maintenance personnel. Operations department will issue this event as required reading to operations personnel. Finally, the Planning Department will review this event and assure that sampling requirements for the Demineralized Water System are included in appropriate maintenance documents. These actions will be complete by November 30, 1997.

Date When Full Compliance was Achieved

Full compliance was achieved on April 14, 1997 when the downstream valve was tagged closed.

NRC Violation 1.d

NRC Regulatory Guide 1.33, Appendix A (8.b.2.aa) recommends procedures for activities, including those for area radiation monitor calibrations.

Surveillance procedure 621.3.005, "High Radiation Monitor (Reactor Building Isolation) and Area Radiation Monitor Power Supply Calibration," steps 6.6.14 and 6.7.14, for reactor building ventilation exhaust radiation monitors A-1 and A-2, respectively, require that IF Upscale or Downscale Trip Points are not within the tolerances specified (13 mR/hr), THEN adjust the trip points.

Contrary to the above, on January 22, 1997, while performing surveillance procedure 621.3.005, instrument technicians miscalibrated both the A-1 and A-2 reactor building ventilation exhaust radiation monitor upscale trip setpoints to 30 mR/hr and 40 mR/hr, respectively.

GPUN Reply To Notice Of Violation 1.d

GPUN concurs with the violation as written.

Reason for the Violation

This violation occurred due to personnel error in that the Instrument and Controls Technician misinterpreted the logarithmic scale on the radiation meter as a linear scale and adjusted the setpoint to 30 mr/hr instead of the setpoint 13 mr/hr. This action was not in accordance with accepted craft practice, or management expectations of expertise. A second similar event occurred on July 9, 1997.

Corrective Steps Taken and Results Achieved

On both January 22, 1997, and July 9, 1997, the Reactor Building ventilation exhaust monitors were reset to the proper values.

In response to the first event, the technicians involved were in the process of conducting a Peer Review training session with emphasis on self checking and having a questioning attitude. Additionally, refresher training on logarithmic scales was being conducted. The critique of this event was included in the Industrial Events Training for Instrument and Control technicians.

Corrective Actions to be Taken to Prevent Recurrence

In response to the July 9, 1997, event, the following corrective actions are being evaluated:

1. The surveillance procedure and the performance of the technicians during these events will receive a full root cause evaluation per site procedures.
2. Additional training on self checking during meter reading and pre job briefings will be conducted.
3. A formal program of disqualification/remedial training/requalification will be implemented in the Maintenance Department.
4. The setpoint of 13 mr/hr will be evaluated for a change to 10 mr/hr to eliminate the concern with interpreting a logarithmic meter.

Date When Full Compliance was Achieved

Full compliance was achieved in both events on the days when the setpoints were properly set.

NRC Violation 2

10 CFR 50.59 requires, in part, that the licensee maintains a record of changes in facility, to the extent that these changes constitute a change as described in the safety analysis report. These records must include a written safety evaluation, which provides the basis for determination that a change does not involve an unreviewed safety question.

Contrary to the above, the licensee made a change to the updated final safety analysis report in 1989, regarding a single failure vulnerability of the standby gas treatment system with regard to automatic actuation, which reduced filter efficiency due to loss of system heaters. However, since 1989, the licensee did not perform, and therefore did not have a record of, a written safety evaluation, which provided the basis for the determination that the change did not involve an unreviewed safety question.

GPUN Reply To Notice Of Violation 2

GPU Nuclear agrees with the material facts presented by the NRC in discussing the circumstances involved with the violation, however, GPU Nuclear denies that a violation of 10 CFR 50.59 occurred.

As stated in the NRC review, the updated final safety analysis report (UFSAR) was changed to correct an error that had been inadvertently introduced in the previous updating process and added the results of an analysis that had been requested by the NRC. A written safety evaluation was not performed since there was no change to the facility (including its licensing basis) or to its procedures. 10 CFR 50.59(b)(1) only requires records of safety evaluations when a licensee makes a change to the facility or its procedures (and then, only if the change to the facility or its procedures affects the accuracy of the safety analysis report). The updating of the UFSAR was made pursuant to 10 CFR 50.71(e).

The standby gas treatment system (SGTS) licensing basis was reviewed with the NRC inspection staff and the results were documented in NRC Inspection Reports 50-219/89-06 and 89-09. The reports concurred with GPU Nuclear that the original licensing basis did not include consideration of single failures. The NRC staff further requested that GPU Nuclear provide justification that the SGTS could be manually started, during a design basis accident, without exceeding 10 CFR Part 100 Exclusion Boundary Dose Requirements. GPU Nuclear conservatively analyzed a manual system start in response to a loss of coolant accident. The results showed offsite doses remained well within 10 CFR Part 100 exclusion boundary dose guideline values. See section 4.0 of Inspection Report 89-09.

In its 1989 UFSAR update, GPU Nuclear fulfilled a commitment to correct the UFSAR so it accurately described the SGTS to reflect the facility's design and licensing basis as documented in the FDSAR. GPU Nuclear also included the results of the manual start analysis. This was done in accordance with 10 CFR 50.71(e) which requires FSAR updates to include "...all analyses of new safety issues performed by or on behalf of the licensee at Commission request."

10 CFR 50.71(e)(2) states that an FSAR update "...submittal shall include (i) a certification by a duly authorized officer of the licensee that either the information accurately presents changes made since the previous submittal, necessary to reflect information and analyses submitted to the Commission or prepared pursuant to Commission requirement, or that no such changes were made; and (ii) an identification of changes made under the provisions of § 50.59 but not previously submitted to the Commission. § 50.71 (e)(2) makes it clear that FSAR update changes can be initiated as a result either of previous information submitted or prepared at Commission request, or of a change, test or experiment evaluated under § 50.59.

In this case, the information describing SGTS design bases had been previously submitted in the FDSAR (the UFSAR predecessor) and provided to the staff in response to the NRC's requests during the 1989 inspection. Moreover, neither the additional analysis of the consequences of a manual start, nor the correction of the UFSAR's general description of ESF initiation features to accurately reflect the original licensing basis as documented in the FDSAR, constituted a change to the facility. There was no physical change to the facility, no change to its design or licensing basis, and no change to its procedures. Therefore, the updating was appropriate under § 50.71(e) and did not require any further review under § 50.59.

As an additional matter, section E2.1.b of NRC Inspection Report 97-03 states that the FSAR change made in 1989 is in conflict with the Technical Specification (TS) basis in Section 4.5 in that limits for containment leakage were based on 90% charcoal filter efficiency. The report further states that since the single failure vulnerability could reduce filter efficiency to 78%, "...the TS basis, specifically the basis for primary containment leakage limits, must consider the limiting single failure vulnerability in order to assure public health and safety." The TS basis in Section 4.5 considers the SGTS as originally designed and licensed. The Section 3.5 (limiting conditions for operation for the containment) basis for the SGTS refers to the FDSAR, Volume 1, Section V-2.4 for a description of the SGTS. In subsection 2.4.1.2 it is stated that "[T]wo separate filter trains are provided, each having 100% capacity. Either of the two filter trains is considered as an installed spare, with the remaining one capable of the required flow capacity." Further, it is stated that "[T]he system is designed for high reliability. To assure system functional availability during any mode of plant operation, critical components are provided with installed spares, including fans, motors, filters and radiation monitors." As can be seen from the FDSAR discussion, the system was designed to be reliable but was not designed to accommodate all single failures. Therefore, the TS basis is not incorrect in assuming a 90% SGTS charcoal filter efficiency. It relates to a reliable system designed to provide an accident mitigation feature that, if needed, would most likely perform its intended function.

As a result of our review of this violation, additional information not previously discussed with the NRC inspectors prior to the issuance of the inspection report was identified. The NRC Safety Evaluation Report (SER-NUREG 1382) for the Oyster Creek full-term operating license (FTOL) evaluated the SGTS single failure vulnerability in Section 6.10. Based on the NRC staff evaluation contained in Inspection Report 89-09, this SER concluded the following:

- The SGTS automatic start logic was not originally designed to meet single failure criteria.
- Loss of power to the reactor building ventilation and filter bank heating coils would not stop the SGTS from performing within its design basis.
- The SGTS could be manually started during a design basis accident without exceeding 10 CFR Part 100 exclusion boundary dose limits.
- The staff found the system adequate.
- The issue was resolved.

The FTOL SER dated January 1991 considered the SGTS evaluation documented in Inspection Report 89-09 a merits review of the single failure vulnerability. The FTOL SER confirms that the requested analysis was adequate to assure SGTS operability (i.e. capability to perform within its design basis). GPU Nuclear considers this to be consistent with its position that the FSAR was appropriately updated in conformance with 10 CFR 50.71(e) and that no violation of § 50.59 occurred. This position is also consistent with industry guidance in effect at the time.

NSAC 125, "Guidelines for 10 CFR 50.59 Safety Evaluations," dated June 1989, states that when the safety analysis report must be changed to bring it into agreement with the NRC approved plant design, the process of updating is addressed in 10 CFR 50.71 (see Section 4.1.1, page 4-4). The NRC did not disagree with this guidance. Further, the most recent industry guidance contained in NEI 96-07 supports the position that a 10 CFR 50.59 evaluation is not required when a facility change is not involved. NEI 96-07 provides examples of activities which do not require a 10 CFR 50.59 evaluation including "...clarifications with no change in the described system function, information already directly approved by NRC, correcting inconsistencies in the SAR and minor corrections to drawings." (See Section 3.12)

The cover letter forwarding Inspection Report 97-03 and this notice of violation asks GPU Nuclear to address in this violation response the GPU Nuclear position regarding the applicability of 10 CFR 50.59. Specifically, the cover letter states that GPU Nuclear's "...current position regarding the need to perform a 10 CFR 50.59 safety evaluation when an original design configuration is subsequently found to not meet all assumed design criteria is inconsistent with the NRC's position that such a configuration necessitates a 10 CFR 50.59 safety evaluation." GPU Nuclear's position is not inconsistent with the staff's expectations. For conditions found which are non-conforming with the design basis and which cannot be immediately restored to meet the design basis, it is GPU Nuclear's policy to evaluate such a condition by documenting a written safety evaluation in accordance with company safety review procedures and 10 CFR 50.59. The issue that is the subject of this notice of violation is not such a case since the condition is in conformance with the design and licensing bases.