U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-219/87-33

Docket No. 50-219

License No. DRP-16 Priority -- Category C

Licensee: GPU Nuclear Corporation
1 Upper Pond Road
Parsippany, NJ 07054

Facility Name: Oyster Creek Nuclear Generating Station

Inspection At: Forked River, New Jersey

Inspection Conducted: October 5 - November 15, 1987

Participating Inspectors: W. H. Bateman, Senior Resident Inspector
W. H. Baunack, Project Engineer, RPS 1A
J. F. Wechselberger, Resident Inspector

Approved by: C J Cowgill III 2/1/88
C. J. Cowgill, Chief, Reactor Projects Section 1A date

Inspection Summary:

Areas Inspected: Routine inspections were conducted by the resident inspectors and one region-based inspector (240 hours) of activities in progress including operations, radiation control, physical security, surveillance, and outage activities. The inspectors also periodically toured the control room and other portions of the plant, reviewed periodic and special reports, observed portions of a quarterly emergency drill, and observed outage related activities. In addition, the inspectors paid particular attention to (1) plans for repairing a pressure seal leak associated with feedwater isolation valve V-2-35, (2) licensee investigations into newly identified areas of drywell thinning at higher elevations in the drywell, (3) identification and correction of an original design error that could have detrimentally affected the ability to achieve secondary containment under certain conditions, and (4) restart related events and activities. Other areas reviewed included "C" main station battery disconnected event, and a review of concerns reported by an instrumentation and control technician.

Results: No violations were identified. Unresolved items are identified in paragraphs 10 and 15 concerning bus undervoltage relay circuitry and the use of short forms in conducting maintenance activities on safety related equipment. Paragraph 15 also discusses incomplete data submittal in response to Bulletin 86-02. The quarterly emergency drill involved a key change in the location of the emergency director from the control room to the Technical Support Center. Licensee plans to freeze seal two ten inch diameter feedwater lines just upstream of their entrance into the reactor vessel were undergoing a rigorous safety review. Evidence of additional drywell shell thinning in certain limited locations at elevations 51' and 82' were undergoing further investigation to determine the extent and cause. The inspectors and regional managers met with the licensee to address pre-startup concerns.

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DETAILS

1. Review of Periodic and Special Reports

Upon receipt, periodic and special reports submitted by the licensee pursuant to Technical Specification requirements were examined by the inspectors. This review included the following considerations: the report includes the information required to be reported to the NRC; planned corrective actions are adequate for resolution of identified problems; and the reported information is valid.

The following reports were reviewed:

-- Monthly Operating Report for September 1987 - Special Report 87-06 dated 10/29/87 involving non-functional fire barrier penetrations as follows:

(1) Two in the east wall of the new cable spreading room due to lack of complete grouting.

(2) One in the ceiling of the 480 volt room due to an opening from an abandoned conduit.

(3) One in the wall between the condenser bay and the feedpump room due to lack of complete grouting.

Fire watches were established in accordance with the Tech Spec requirements until repairs were made.

2. Drywell Shell Thinning

During the report period, several meetings between the licensee and the NRC resulted in the licensee extending their investigation to determine a more definite boundary to the thin locations identified during planned ultrasonic depth inspections of the shell at elevations 51' and 82'. The licensee also elected to take core samples of the shell in certain of the affected areas in an attempt to (1) confirm the ultrasonic readings and (2) determine the mechanism causing the thinning. Specific results of these investigations are documented in NRC Inspection Report 87-38. The licensee's overall conclusion, based on the calculations performed using the results of the drywell shell thickness inspections, was that the drywell was sufficiently structurally sound to perform its design function. Licensee efforts are continuing to identify the source of water involved in the thinning action and to determine methods to stop the corrosion. The inspectors had no further concerns at this time.

3. Radiation Protection

During entry to and exit from the RCA, the inspectors verified that proper warning signs were posted, personnel entering were wearing proper dosimetry, personnel and materials leaving were properly monitored for radioactive contamination, and monitoring instruments were functional and in calibration. Posted extended Radiation Work Permits (RWPs) and survey status boards were reviewed to verify that they were current and accurate. The inspector observed activities in the RCA to verify that personnel complied with the requirements of applicable RWPs and that workers were aware of the radiological conditions in the area.

A question was raised by licensee personnel as to the accessibility of the drywell beneath the space bounded by the drywell head. This question was asked during a radcon briefing prior to making an entry into the drywell head space through an open manway on the head. The concern was if the lower levels of the drywell were accessible, then the manway opening should be treated as an access point to a locked high radiation area. The question was also raised as to whether or not the radiation levels in the reactor cavity met locked high radiation criteria. Surveys and inspections were performed and it was determined that the radiation levels in the reactor cavity area and under the drywell head were not high enough to meet the criteria for a locked high radiation area. Additionally, it was determined that access to lower levels of the drywell from the space underneath the drywell head was not possible based on the largest width of any opening being 6". The NRC inspectors questioned licensee personnel in detail about the accessibility because, in the past, access to the drywell head area from lower elevations in the drywell was necessary to operate the reactor head vent valves. The licensee explained to the inspectors that the space available for this access had been taken up by conduit used to supply power to new electrically operated reactor head vent valves installed during the recent 11R outage. The inspectors had no further questions about the radiological aspects of this issue.

Two radiological incidents were reported to the NRC inspectors during this report period. One involved workers improperly entering the drywell which is a locked high radiation area, and the other involved two separate occasions when workers entered the drywell without proper dosimetry. NRC Inspection Report 87-39 discusses the details of these incidents. Another incident reported during this report period and discussed in the above inspection report involved identification by licensee personnel of broken cage wire next to the lock of the locked high radiation door controlling access into the reactor water cleanup area. The break was such that an individual could reach through the hole and unlock the door from the inside. The licensee investigated and determined that the probable cause for the broken wire was a result of cyclic fatigue from extended use of the wire on the cage door as a hand hold. The licensee placed sheet metal over the area to repair the area and restore the door security. Subsequent review of TLD data did not indicate any unexplained high radiation exposure.

4. Observation of Physical Security

During daily tours, the inspectors verified that access controls were in accordance with the Security Plan, security posts were properly manned, protected area gates were locked or guarded and that isolation zones were free of obstructions. The inspectors examined vital area access points to verify that they were properly locked or guarded and that access control was in accordance with the security plan.

10 CFR 73.55 discusses requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage. Paragraph 73.55 (d) (8) requires that access into containment be controlled and states the options for instituting this control. A question regarding proper security measures with regards to these requirements was identified by the licensee. NRC Inspection Report 50-219/87-40 discusses this issue in greater detail. Once the situation was identified to site security, proper compensatory measures were taken.

The inspectors had no additional concerns.

5. Drywell Purge Valves-Lack of a Design Feature to Automatically Close

Licensee pursuit of a Preliminary Safety Concern resulted in identification of an original plant design deficiency involving operation of valves in the drywell purge line. When purging the drywell, reactor building supply fans SF 1-12, 13, and 14 supply air through a flow path that contains four air operated butterfly valves. Two of these valves, V-27-3 and V-27-4, automatically close on reactor vessel low low water level or drywell high pressure. The other two, V-28-42 and V-28-43, do not have any auto close feature. All four valves can be operated from the Control Room. Upon receipt of a high-high radiation signal in the reactor building ventilation exhaust or on the refueling floor, supply fans SF1-12, 13, and 14 auto stop, an auto start signal initiates the standby gas treatment (SBGT) system, and reactor building isolation occurs to achieve secondary containment. In this condition the Tech Specs require that the SBGT system be able to achieve a negative 1/4" water pressure in secondary containment relative to barometric pressure outside the reactor building. Because none of the four valves in the purge line close when the high high radiation signal is received and the SBGT system takes suction from the drywell as well as the reactor building, a flow path exists directly from the outside through the loosely closing dampers on the discharge side of the supply fans, past the four open valves and into the drywell. The existence of this flowpath gave rise to the safety concern that a negative pressure in the secondary containment may not be achieved.

The licensee informed the NRC inspectors of this design deficiency. The licensee and the inspectors agreed that a test should be run to determine whether or not a negative 1/4" H_2O differential pressure could be achieved in the above described lineup. The test was performed and the results indicated that the concern was valid since only a very small negative differential pressure was achieved. At this point the licensee and the NRC inspectors discussed actions that should be taken to eliminate the problem scenario. Operations personnel agreed to establish administrative controls such that the drywell would not be purged when secondary containment, by Tech Spec definition, was required. Engineering support personnel commenced designing a modification that would make V-28-42 and 43 automatically close upon receipt of a high-high radiation signal from either the reactor building ventilation exhaust or the refueling floor.

Prior to completion of the 11 M outage, design, installation, and testing of the modification were completed. The inspectors had no further questions. The NRC inspectors considered that the identification of this design deficiency and its prompt and thorough technical resolution were indicators of more aggressive engineering support.

6. Plans to Repair Feedwater Isolation Valve V-2-35

One of the major repair activities of the 11 M outage was the leak past the pressure seal of feedwater isolation valve V-2-35. This valve is located in the drywell and the leak was a major contributor to the drywell unidentified leakrate prior to the shutdown. Several repair schemes were considered including a freeze seal option that was eventually selected.

Because the feedwater line downstream of V-2-35 is unisolable from the reactor vessel any repair scheme involving disassembly of the valve would require either defueling the core and partially draining the vessel or establishing a blockage in the line between the valve and the reactor vessel. The 18 inch feedwater line splits into two 10 inch lines downstream of V-2-35, and the freeze seal option proposed to establish a freeze seal in both of the 10 inch lines. The NRC inspectors were concerned about the potential for serious consequences should the freeze seal fail when the valve was open for repair. In a comprehensive safety evaluation, the licensee addressed all possible scenarios that could lead to problems and planned provisions for reacting to the unexpected.

As part of the planning effort, a mockup freeze seal was established in a 10 inch piece of piping similar in material properties to the feedwater piping. This mockup freeze seal afforded the licensee an opportunity to verify the seal could be established. Once established a hydrostatic pressure was applied to one side of the freeze seal to demonstrate that the ice plug could not be displaced. Other tests and measurements were also made with results indicating the freeze seal was a viable option.

The valve was repaired using the freeze seal to provide isolation from the reactor vessel without incident. The inspectors reviewed the safety evaluation in detail, witnessed the mockup testing, and followed activities of the onsite freeze seal and valve repair activities. This activity was a major effort involving mainly the engineering support and maintenance groups. The overall success of the effort demonstrated the ability of the licensee to successfully perform a complicated evolution involving effective, interdepartment communication and coordination. No inspector concerns were identified.

7. Review of Strike Plans

The IBEW union contract expired at the end of October. Negotiations were in progress at the end of October but no agreement had been reached, and the union agreed to continue to work without a contract for an undefined period of time. Because of the threat of a strike, the NRC inspectors reviewed the licensee's plans for coping with a strike should one occur. Although the plant was shutdown, which required a minimum complement of licensed operators, restart from the 11 M outage was imminent, thereby requiring a full complement of licensed operators. A review of the proposed watch bill indicated that a sufficient number of licensed operators were available to meet Tech Spec manning requirements with some control room watch standing required for a few individuals to re-establish their qualification. This action was accomplished. Provisions for relieving the watch if a strike occurred and provisions for eating and sleeping were also reviewed. Several discussions between licensee and NRC Region management resolved all outstanding questions and requirements. The NRC determined the licensee was adequately prepared to cope with a strike. [Note added, Inspection Report 50-219/87-41 details activities during the strike.]

8. 11 M Outage Activities

The inspectors observed various activities during the 11 M outage. Included were:

-- local leakage rate testing of containment penetrations.

-- integrated leakage rate testing of containment; NRC Inspection Report 87-35 details results of the containment integrated leakage rate test.

-- NDE of the reactor cavity liner and the bellows seal between the drywell and the bottom of the reactor cavity; NDE inspections of these areas identified a substantial number of indications at or adjacent to the welds that will require repair prior to the next refueling. A portion of all of the leakage into the air gap between the biological shield wall and the drywell may be through some of these weld defects.

-- replacement of 5 electromatic relief valve acoustic monitor splices with a new design splice.

-- reactor building roof repairs; repairs to the reactor building roof have identified areas of water pockets underneath the roofing material and rusted Q-decking as a result. These areas are the most likely contributors to rainwater inleakage into the reactor building.

-- main flash tank manway repairs; a new design manway cover and newly machined seating surfaces ensure a final fix of the chronic problem with leaking main flash tank manways.

-- "A" recirculation pump seal replacement.

-- installation of the 600 psig scram reset modification.

-- replacement of failed intermediate range detectors; these continued to fail with no obvious explanation. Investigations as to the cause were continuing at the end of the report period.

-- drywell mechanical snubber inspections; no defective mechanical snubbers were identified in the drywell. Approximately 90 snubbers were inspected.

-- inspections and repairs to the drywell to torus vacuum breakers.

The inspector had no other concerns in these areas.

9. Cold Weather Preparations

During routine tours of the site, the inspectors looked at the condition of heat tracing and insulation on outdoor safety-related piping systems and toured outlying buildings where, in the past, freezing problems had been experienced. Several deficiencies were observed including non-functional heat tracing, broken and thereby ineffective insulation, and missing insulation. The inspectors noted that these were similar to problems that have existed in the past and expressed their concerns to the licensee who stated that due to the higher priority outage works, efforts would be directed to readying the plant for cold weather as work crews became available after completion of outage related items. The inspectors will continue to follow licensee activities in this area.

10. "C" Main Station Battery Disconnected

On October 30, 1987, the licensee discovered that the "C" main station battery (125 VDC) and the #2 emergency diesel generator (EDG) were both inoperable during the period October 13-19, 1987. The "C" battery supplies control power to #1 EDG, 4160 V buses 1C and 1A and 460 V 1A2 bus. In this situation with a complete loss of offsite power from the available four sources, no emergency diesel generator would be automatically available to supply power to the emergency 4160 volt buses 1C and 1D. Therefore, the required safety systems for the plant's cold shutdown condition, core spray and standby gas treatment systems would not have automatically initiated if required by an accident condition coupled with a complete loss of offsite power.

The "C" battery was taken out of service for an equalizing charge, test discharge and subsequent equalizing charge on 9/10/87. The battery breaker remained open until 10/30/87 when it was returned to service. A static charger had been supplying 125 VDC power while the "C" battery breaker was open. On 10/13/87 the #2 EDG was removed from service for a maintenance inspection and returned to service on 10/19/87. The delay experienced in returning the battery to service resulted from difficulties associated with the battery charging.

On 11/9/87 the site inspectors and regional management met with the licensee to discuss this event as well as other issues (see paragraph 13). The licensee described the event and outlined the corrective action to be taken to prevent recurrence. The inspector questioned the licensee if they had examined this time period to determine if either the core spray system or the standby gas treatment system were out of service for maintenance other than the 10/13-19/87 period. The licensee later examined available information to determine that these systems were not removed from service. In addition the inspector questioned the appropriateness of the design of the undervoltage relay circuitry for the 4160 V emergency and the 460 V vital buses. These relays have to be energized to actuate on an undervoltage condition. The "C" battery would be required to supply power to these relays during a loss of offsite power condition, but in this situation this would not occur and the undervoltage condition on the bus would not be sensed. Therefore the design is not a fail safe design. The licensee developed a similar concern and reviewed this condition. This item will remain unresolved pending further inspector review (87-33-01).

The inspector concluded that the event had a low probability of occurrence and that the safety significance was minimal as the plant was in a cold shutdown condition. It is significant that required safety systems would be unable to function as designed, but considering plant conditions, the event's safety significance was considered minimal. The licensee completed corrective action to prevent occurrence which will be examined with the licensee event report review.

11. Safety Valve Release Nut

On 11/10/87 while the inspector was witnessing the hydrostatic test conducted to ensure the integrity of V-2-35, operations personnel detected high unidentified leak rate greater than 10 gpm on the control room recorder. Personnel immediately entered the drywell to determine the source of the leak and found safety relief valve V-28-F not fully seated and issuing water. The hydrostatic test was secured while the licensee determined the cause of and corrective action to be taken to repair the leaking safety relief valve.

Upon close examination of the V-28-F, the licensee discovered that the release nut cotter pin had broken allowing the release nut to move down the valve stem as a result of system flow induced vibration. The nut came to rest on the valve yoke, causing the valve stem to raise on the subsequent plant cooldown and shutdown. The inspector reviewed the valve mechanics with the licensee to ensure that this did not affect safety relief valve setpoint actuation and that the removal of the release nut would not affect valve performance. The licensee elected, with the manufacturer's concurrence, to remove all the release nuts from the 16 safety relief valves. The release nut is used in conjunction with a drop lever to manually actuate the valve if that is desirable but at Oyster Creek it serves no purpose. In addition, V-28-F was replaced with a spare safety relief valve which was first appropriately tested by a contract laboratory for the correct lift setpoint. During the subsequent hydrostatic test, the safety relief performed as designed without encountering any significant seat leakage.

The inspector had no further concerns.

12. Plant Operation Review

12.1 Routine tours of the control room were conducted by the inspectors during which time the following documents were reviewed:

-- Control Room and Group Shift Supervisor's Logs:

-- Technical Specification Log;

-- Control Room and Shift Supervisor's Turnover Check lists;

-- Reactor Building and Turbine Building Tour Sheets;

-- Equipment Control Logs;

-- Standing Orders; and,

-- Operational Memos and Directives.

12.2 Routine tours of the facility were conducted by the inspectors to make an assessment of the equipment conditions, safety, and adherence to operating procedures and regulatory requirements. The following areas are among those inspected:

-- Turbine Building

-- Vital Switchgear Rooms

-- Cable Spreading Room

-- Diesel Generator Building

-- Reactor Building

The following additional items were observed or verified:

a. Fire Protection:

-- Randomly selected fire extinguishers were accessible and inspected on schedule.

-- Fire doors were unobstructed and in their proper position.

-- Ignition sources and combustible materials were controlled in accordance with the licensee's approved procedures.

-- Appropriate fire watches or fire patrols were stationed when equipment was out of service.

b. Equipment Control:

-- Jumper and equipment mark-ups did not conflict with Technical Specification requirements.

-- Administrative controls for the use of jumpers and equipment mark-ups were properly implemented.

No unacceptable conditions were identified.

13. Onsite Plant Restart Meeting

On 11/9/87 regional managers and the site inspectors met with the licensee to discuss issues relating to restart of the plant. The following issues were discussed: (1) the local leak rate surveillance for containment isolation valves (see paragraph 15)., (2) "C" main station battery (125VDC) and the #2 emergency diesel simultaneously out of service (see paragraph 10), (3) security access to containment (see Inspection Report 87-37), (4) high radiation barrier to the reactor water cleanup room degradation (see paragraph 3), and (5) drywell wall thinning concerns (see paragraph 2.0).

The licensee addressed the inspectors' concerns in their presentation.

14. Quarterly Emergency Drill

The inspector observed portions of the quarterly emergency drill on 11/4/87. Drill activities were observed in the control room and the technical support center. The inspector concluded that the drill was adequately performed and demonstrated the shifts' ability to assess and effectively operate the plant under emergency conditions. Emergency plans and procedures were carried out in accordance with the requirements.

This drill involved a change in the emergency director's location from the control room to the technical support center. The inspector observed the emergency director to be effective in his new location.

15. Instrumentation & Control Concerns

On 10/30/87 a worker at the plant approached the inspectors with a number of concerns. These included on the job training irregularities, lack of a meaningful cyclic instrumentation and control (I&C) training program, supervisor instructions not to talk to the NRC or Institute of Nuclear Power Operations representatives, required overtime to perform housekeeping duties, or no work at all, local leak rate tests (LLRT) are repeated until satisfactory results are achieved, technicians are instructed to perform surveillances or maintenance actions with an 'inadequate' procedure, technicians were chastised for following procedures while performing a surveillance, and low department morale as a result of management practices. With the concurrence of the individual the inspectors briefed licensee management on these concerns and requested that they be addressed. In addition, the inspectors elected to independently review some of the issues in addition to the licensee's efforts. Additionally, the inspectors chose to independently review the results of LLRTs conducted on six containment isolation valves and the circumstances surrounding a surveillance conducted on a reactor building to torus vacuum breaker differential pressure switch (DPS 66B). It was requested that the licensee present the results of their investigation to the inspectors and Region I management (see paragraph 13).

The inspectors reviewed the events surrounding the reactor building-to-torus vacuum breaker differential pressure switch (DPS-66B) surveillance (604.3.001) conducted on 10/28/87. During the surveillance the technician could not achieve the proper reset point on the instrument and stopped the surveillance, as the procedure was not clear in its guidance if a reset point was out of specification. Also, the procedure did not require "as left" and "as found" reset point data. The reset point is not critical to the proper performance of the safety related reactor building-to-torus vacuum breakers but does indicate a potential instrument problem. The differential pressure switches in question are manufactured by Static-O-Ring. Inspection Reports 86-02, 86-04, and 86-06 provide more detail on Static-O-Ring differential pressure switches.

Subsequently, the licensee investigated the apparent failure and had other technicians complete the surveillance on the DPS-66B instrument. Additionally, they requested the technicians to perform a second surveillance of the instrument for Short Form 47682 (Job Order); which is an extended short form for minor maintenance. During this surveillance a proper reset point within the specified tolerance was achieved. Subsequent to these surveillances and the concerns raised by technicians, the licensee performed additional testing on the DPS 66A&B instruments and elected to replace DPS-66B with a spare. The inspector reviewed the spare differential pressure switch bench testing conducted by the licensee to ensure the spare switch did not exhibit the same anomalous characteristics as the installed switch. The bench testing data showed repeatability for the trip setpoint and reset point.

To clarify the surveillance procedure the licensee initiated a revision to provide additional guidance on reset point acceptability. The inspector reviewed the revised surveillance procedure. The licensee incorporated steps to record the "as found" reset point. In addition, steps for three "as found" and "as left" trip and reset point data requirements were added.

During the review of this surveillance the inspector noted that work was performed on DPS-66B under an extended short form (47682) which does not receive the same review as a short form. DPS-66B is a nuclear safety related and environmentally qualified equipment which should require the review received when safety related work is processed under a short form. In addition, the inspector noted instances where the valve verification checkoff is performed by the same individual assigned to perform the surveillance which is apparently contrary to the procedure requirement to have an individual perform the valve verification who did not witness the initial valve lineup. These items are unresolved pending further inspector review (87-33-02). In addition, on one surveillance sheet the inspector noted an almost twelve hour delay from the time testing was completed until the group shift supervisor (GSS) reviewed and signed the surveillance review form. This is significant in this case as the surveillance was noted to have a discrepancy and paragraph 4.3 of the governing surveillance (604.3.001) requires the GSS to be notified immediately.

The inspectors reviewed the LLRT data sheets on selected valves in reactor water cleanup system and reactor building closed cooling water system to determine that the tests were conducted properly in accordance with licensee procedures. The following valve data sheets were reviewed: V-16-1, V-16-2, V-16-14, V-16-61, V-5-166, V-5-167 and V-5-147. V-5-147 showed LLRT failures on 10/10 and 10/20/87 and subsequent repairs and final test and acceptance on 10/22/87. The inspectors' review found the LLRT's to be conducted in accordance with the licensee's procedures and failures to be documented, corrective action taken and final test and acceptance performed. The inspectors had no further questions regarding the performance of local leak rate testing.

During the review of the DPS-66 surveillances the inspector reviewed the licensee's response to Bulletin 86-02: Static "O" Ring Differential Pressure Switched which required the submittal of information on Static-O-Ring instruments used in safety related plant applications. In response the licensee submitted a letter dated 9/23/86 to the Region I administrator which included surveillance calibration data on DPS-66A& B. This data was supplied to support their response to item 6 of the bulletin which in part required a report to "describe the long term corrective actions to be taken, including the implementation schedule, the impacts of potential common mode failures, and an analysis to demonstrate that the system involved will meet regulatory requirements and function reliably".

In reviewing this submittal against current equipment history data the inspector noticed a discrepancy in the reported data. The following data was not submitted in the bulletin response:

DPS-66A
Date: 8/29/86

Trip Setpoint (11.07 ± 1.38" W.G.)	Reset Point (7.07 ± 1.5" W.G.)
As found 12.8	10.0
As left 10.4	10.0

DPS-66B
Date: 7/24/86

As found 13.3	5.6
As left 11.0	5.6

Date: 8/29/86

As found 11.4	10.4
As left 11.4	10.4

The technical specification requires the trip setpoint to be less than or equal to 13.84" W.G. which was met in all cases. Apparently this was an inadvertent oversight between the onsite and the offsite engineering organizations. Presently the licensee is reviewing the surveillance calibration data to determine if the inclusion of the missing data affects the original licensee bulletin submittal. The licensee will provided the results of the review to the NRC.

Currently the licensee is continuing their investigation into I&C technicians' concerns. Regarding the worker concerns related to not talking to the NRC and INPO, initial licensee investigation concluded that the worker misunderstood a supervisor's comments with regard to speaking to the agencies. However, the licensee re-emphasized their long standing commitment for open communication and counselled the supervisor to strive for more effective communications. The inspectors are closely following this effort and will review the licensee's investigation when it is complete.

16. Unresolved Items

Unresolved items are matters for which more information is required in order to ascertain whether they are acceptable, violations, or deviations. Unresolved items are discussed in paragraphs 10 and 15 of this report.

17. Exit Interview

A summary of the results of the inspection activities performed during this report period were made at meetings with senior licensee management at the end of this inspection. The licensee stated that, of the subjects discussed at the exit interview, no proprietary information was included.