U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-440/88015(DRP)

Docket No. 50-440 License No. NPF-30

Licensee: Cleveland Electric Illuminating Company
Post Office Box 5000
Cleveland, OH 44101

Facility Name: Perry Nuclear Power Plant, Unit 1

Inspection At: Perry Site, Perry, Ohio

Inspection Conducted: August 24 through October 18, 1988

Inspectors: K. A. Connaughton

G. F. O'Dwyer

Approved By: R. W. Cooper
Richard Cooper, Chief
Reactor Projects Section 3B

11/10/88
Date

Inspection Summary

Inspection on August 24 through October 18, 1988 (Report No. 50-440/88012(DRP))
Areas Inspected: Routine, unannounced inspection by resident inspectors of previous inspection items, Temporary Instruction (TI) 2515/81, "Inspection Requirements For IE Bulletin 86-02, 'Static "O" Ring Differential Pressure Switches,'" TI 2515/82, "Inspection Requirements For IE Bulletin 86-01, 'Minimum Flow Logic Problems That Could Disable RHR Pumps,'" TI 2515/93, "Inspection for Verification of Quality Assurance Request Regarding Diesel Generator Fuel Oil Multi-Plant Action Item A-15," NRC regional office requests, operational safety, nonroutine events, and surveillance testing. NRC and licensee management met on August 24 and September 30, 1988 to discuss licensee performance and recent operational events.
Results: Of the nine areas inspected, one violation was identified in one area (failure to notify the NRC of inoperability of both trains of the control room emergency recirculation system within the timeframe required by 10 CFR 50.72 - Paragraph 6.c.). On September 4 and 16, 1988, offgas hydrogen burns occurred following offgas system transients which resulted in quenching of the hydrogen recombiners. Initial inspector response to these events was performed by the resident inspectors. Detailed followup inspection was performed by a special inspection team from the NRC Region III Division of Reactor Safety. The results of the special team inspection will be documented in NRC Inspection Report 440/88016(DRS).

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DETAILS

1. Persons Contacted

   a. Cleveland Electric Illuminating Company

      #+Alvin Kaplan, Vice President, Nuclear Group
      +C. M. Shuster, Director, Nuclear Engineering Department (NED)
      *#+M. Lyster, General Manager, Perry Plant Operations Department (PPOD)
      *#+R. A. Stratman, Manager, Operations Section, (PPOD)
      # M. Wesley, Acting Senior Operations Coordinator (PPOD)
      #+V. K. Higaki, Manager, Outage Planning Section (PPOD)
      # F. R. Stead, Director, Nuclear Support Department (NSD)
      *# W. R. Kanda, Manager, Instrumentation and Controls Section (PPTD)
      *+#S. F. Kensicki, Director, Perry Plant Technical Department (PPTD)
      L. L. Vanderhorst, Radiation Protection Section (PPTD)
      *#+R. A. Newkirk, Manager, Licensing and Compliance Section (NSD)
      #+K. Pech, Manager, Technical Section (PPTD)
      +E. Riley, Director, Nuclear Quality Assurance Department (NQAD)
      *#+G. R. Dunn, Compliance Engineer (NSD)
      T. A. Boss, Supervisor, Quality Audit Unit (NQAD)
      * D. J. Takas, Manager, Mechanical Maintenance Quality Section (NQAD)

   b. U. S. Nuclear Regulatory Commission

      +R. C. Knop, Chief, Projects Branch 3
      #+R. W. Cooper, II, Chief, Projects Section 3B
      *#+K. A. Connaughton, Senior Resident Inspector
      *#+G. F. O'Dwyer, Resident Inspector

   *Denotes those attending the exit meeting held on October 18, 1988.
   +Denotes those attending the August 24, 1988 plant status meeting.
   #Denotes those attending the September 30, 1988 plant status meeting.

2. Licensee Action on Previous Inspection Findings (92701, 92702)

   a. (Closed) Open Item (440/86023-05(DRP)): Provisions did not exist for delegation of the fire brigade command function in the event the fire brigade leader was unavailable and/or incapacitated. The inspectors reviewed Temporary Change Notice (TCN)-1 to Plant Administrative Procedure (PAP)-1911, "Fire Emergency," Revision 2, dated September 14, 1988. The subject TCN provided an expanded description of fire attack leader responsibilities such that if the fire brigade leader became incapable of performing his duties, the fire attack leader was to assume responsibility for the fire brigade until a replacement fire brigade leader is available. In addition, the licensee revised PAP-201, "Conduct of Operations" to ensure that the Supervisory Operator (licensed reactor operator) designated as the shift fire brigade leader was not concurrently assigned duties as the "operator at the controls." The inspectors have no further concerns regarding this matter.

b. (Closed) Open Item (440/87008-01(DRP)): Resolution of discrepancies between reactor recirculation flow and predicted total core flow identified during startup testing. The subject discrepancy was documented on Test Exception Report (TER)-232-2. During this inspection, the inspectors reviewed the licensee's documented resolution of TER-232-2. Startup test data acquired in Startup Test Condition 3 was evaluated in consultation with personnel from General Electric. The evaluation concluded that lower-than-predicted core flows at various conditions of recirculation flow and reactor power level were likely due to higher-than-predicted core plate and other internal loop frictional pressure drops. It was predicted from this data that rated core flow at 100 percent power could be achieved with a 0 to 3 percent excess core flow capability and while no restrictions existed on operation at rated power, some loss of Maximum Extended Operating Domain excess flow was expected. It was recommended that testing proceed through Test Condition 6 for a complete set of reactor recirculation flow, core flow, and core plate differential pressure measurements. Until such data was collected, core flow was to be restricted such that a core plate differential pressure of 24.1 psi was not exceeded.

Additional testing conducted during Test Condition 6 validated predictions based upon the Test Condition 3 data. The core flow shortfall in Test Condition 6 at 100 percent power was approximately 5 percent. General Electric's subsequent evaluation of all the test data concluded that the magnitudes of the identified discrepancies did not impose any additional or new thermal hydraulic restrictions on continued reactor operation.

In order to ensure that recirculation flow elbow tap instrumentation utilized in the APRM flow-biased trip reflected actual system performance, General Electric Field Deviation Disposition Request (FDDR) KL1-6593 was issued. Reanalysis of the recirculation flow/core flow relationship was performed utilizing startup test data. The revised predictions from this analysis were utilized to respan the recirculation flow elbow tap instrumentation to properly reflect equilibrium, end-of-cycle rated drive flow. The disposition of FDDR KL1-6593 was implemented by Design Change Package (DCP)-870514. DCP-870514 was fully implemented on September 23, 1987. The inspectors have no further concerns regarding this matter.

c. (Closed) Violation (440/87012-06(DRP)): Loss of primary containment integrity due to mispositioned manual containment isolation valves 1P54-F726 and 1P54-F727. This violation resulted from a lack of coordination in the performance of procedural actions required by Integrated Operating Instructions (IOI)-4, "Shutdown" and IOI-1, "Cold Startup." In order to prevent recurrence, the licensee revised IOI-1 to include verification that the subject valves are closed immediately prior to declaring that containment integrity has been established. This change ensured that all other IOI's were closed prior to performing the verification. Independent verification of valve closure was also procedurally required.

IOI-4 was revised to provide additional guidance as to when the isolation valves should be opened upon plant shutdown. Specifically, the valves were only to be opened if transient combustible or burn permits requiring fire suppression capability in containment were in effect, or the plant was expected to remain shut down for greater than 48 hours. The inspectors reviewed TCN-18 to IOI-1, Revision 3 and TCN-20 to IOI-4, Revision 2 which implemented the above described procedural changes and found them to be satisfactory.

d. (Closed) Violation (440/87012-07(DRP)): Failure to take timely and effective corrective action for repetitive engineered safety features actuations. Repetitive engineered safety features actuations occurred during electrical distribution system switching operations involving the RPS electrical busses. Additionally, repetitive Reactor Core Isolation Cooling (RCIC) isolations occurred as a result of leak detection system trips without a definitive root cause being identified.

With regard to the ESF actuations associated with the loss of an RPS bus, the licensee performed a detailed engineering review of loads supplied by the RPS busses. Expected responses of the various loads to a loss of power were categorized as; automatic actions, partial trips, loss of a circuit/permissive, loss of a bypass/seal-in/reset, annunciator alarm/out-of-service, process computer alarm/out-of-service, ERIS point actuation/out-of-service, or loss of indication/instrumentation. This evaluation resulted in the development of a load list which included associated failure mode effects. The list was forwarded to licensee operations and technical staff personnel for incorporation into appropriate procedures. The inspectors reviewed System Operating Instruction (SOI)-C71, Revision 6, "RPS Power Supply Distribution (Unit 1)," TCN-9 to Off-Normal Instruction (ONI)-C71-2, Revision 0, "Loss of One RPS Bus (Unit 1)," TCN-4 to Surveillance Test Instruction (SVI)-C71-T5232, Revision 3, "Reactor Protection System-Electrical Power Monitoring Calibration/ Functional for 1C71-S003B and 1C71-S003D," and TCN-4 to SVI-C71-T5230, Revision 3, "Reactor Protection System-Electrical Power Monitoring Calibration/Functional for 1C71-S003A and 1C71-S003C." These procedures were the principle procedures pertaining to either planned or unplanned RPS bus deenergization. The procedures were appropriately revised to incorporate failure mode effects identified by the engineering review which were not previously addressed. The inspectors are satisfied that these and earlier procedure improvements will collectively minimize the likelihood of unexpected ESF actuations during RPS bus switching operations.

In order to address the repetitive RCIC isolations, the licensee implemented a design change which provided indication of RCIC isolation relay status. Isolation relay status could then be easily verified by licensee personnel prior to enabling the isolation function. The inspector verified by direct observation and review of DCP-870666 that the subject modification was completed on February 8, 1988. Based upon subsequent operating history, the inspector is satisfied that this design change has been effective in avoiding unnecessary RCIC isolations upon restoration of leak detection instrumentation following routine surveillances.

e. <u>(Closed) Violation (440/88004-02(DRP))</u>: Misadjustment of APRM gain settings resulting in Technical Specification violation. APRM gain adjustments based upon process computer calculations of thermal power were rendered invalid as a result of a failed sensor input. Self checks by the process computer identified the failed input; however, personnel performing the APRM gain adjustment did not check the process computer output to ensure it's validity. The inspectors reviewed the current revision of SVI-C51-T0024, "APRM Channel Calibration Evaluation/Adjustment" and determined that it had been appropriately revised to require that the process computer output be specifically checked to ensure it's validity prior to making APRM gain adjustments. The inspectors are satisfied that this procedural change will preclude future similar occurrences.

f. <u>(Closed) Open Item (440/88004-03(DRP))</u>: Initially a mechanic improperly connected MOVATS equipment to a valve and had to consult with a Senior Maintenance Technician (SMT) to achieve proper installation. Generic Mechanical Instruction (GMI-0056)," Motor Operated Valve Analysis and Test System (MOVATS) Testing" was revised and training was accomplished in order for all mechanics (certified to perform MOVATS testing) to be able to properly compensate for any differences between the sample wiring diagram and the actual wiring of any valve. The inspectors have no further concerns regarding this matter.

g. (Open) Open Item (440/88012-05(DRP)): Evaluation of Loose Parts Monitoring System (LPMS) alarm condition. During this inspection period, the licensee continued the evaluation of LPMS alarms received between August 8 and 14, 1988. The licensee issued an interim report, dated September 30, 1988, which summarized the extent of evaluations conducted thus far and conclusions regarding the source of the LPMS alarm. Specifically, it was concluded that the LPMS alarm was not attributable to a loose part or component. The alarm was, instead, the result of three contributing sources: flow induced background noise; structural resonance of the recirculation pump and associated piping; and a low frequency noise component on LPMS channel 7. When all contributing signals coincided, an alarm condition would occur. A final report based upon analyses performed by the licensee, General Electric, Gilbert/Commonwealth Inc., and Advanced Technologies LPMS experts was expected to be complete on December 1, 1988. Additionally, the licensee has committed to perform inspections of the reactor internals during the first refueling outage in order to confirm the absence of a loose part. Completion of the ongoing loose part evaluation, issuance of the final report, and inspection of the reactor vessel internals will continue to be tracked by this open item.

3. NRC Inspection Manual Temporary Instructions (TI's)

a. TI 2515/81, "Inspection Requirements For IE Bulletin 86-02, 'Static "O" Ring Differential Pressure Switches.'" (25581)

Inspector reviews required by this Temporary Instruction were perviously accomplished during an inspection documented in NRC Inspection Report No. 440/86023(DRP). This Temporary Instruction is therefor considered closed.

b. TI 2515/82, "Inspection Requirements For Compliance Bulletin 86-01, 'Minimum Flow Logic Problems That Could Disable RHR Pumps.'"(25582)

Inspector reviews required by this Temporary Instruction were previously accomplished during an inspection documented in NRC Inspection Report No. 440/86025(DRP). This Temporary Instruction is therefor considered closed.

c. TI 2515/93, "Inspection For Verification of Quality Assurance Request Regarding Diesel Generator Fuel Oil Multi-Plant Action Item A-15" (25593)

Based upon the vintage of the Perry plant, the licensee was not a recipient of a 1980 letter from the NRC Office of Nuclear Reactor Regulation which requested that licensees either include diesel generator fuel oil in their quality assurance programs or provide a letter of justification for not doing so. The inspectors did, however, determine what measures the licensee has in place to assure diesel fuel oil quality.

The licensee procures No. 2 diesel fuel oil non-safety-related from a commercial supplier. As conditions of the purchase agreement, the supplier was to supply a certificate of conformance with each shipment attesting to the fact that the fuel oil meets the requirements of ASTM standard D-975-77. Additionally, the fuel oil shall meet the following: oxidation stability per ASTM standard D-2274-70, and; 2 mg/100ml API gravity at 60 degrees F. Prior to the offloading of a given shipment, licensee chemistry personnel sample and independently analyze the fuel oil in accordance with the above specification for water, sediment, Saybolt viscosity and API gravity. If the results of the analyses are unacceptable, the entire shipment is rejected. The Perry, Unit 1 Technical Specifications reflect all of the above requirements.

The licensee's chemistry program was, in turn, covered by the licensee's Q.A. program and all relevant 10 CFR 50, Appendix B quality assurance criteria were applicable to diesel fuel oil chemical analyses performed by chemistry personnel.

No violations or deviations were identified.

4. NRC Regional Office Request - Cracking of Welds on TDI Diesel Generator Intercooler Adapter Guide Vanes (92701)

During this inspection period, the inspectors were apprised by NRC Regional Office personnel of a problem experienced at the Grand Gulf nuclear power plant concerning the standby diesel generators. The problem involved the cracking and ultimate failure of welds which secured guide vanes to the diesel intercooler adapter housing. As a result the guide vanes may break loose and impinge upon intercooler cooling coils causing them to leak. This may ultimately result in derating of the diesel generators. The inspectors contacted licensee engineering personnel with responsibility for the Perry diesel generators and determined that they were aware of the Grand Gulf experience.

The inspectors were informed by licensee personnel of a separate but perhaps related problem involving intercooler adapter housing welds. Based upon a vendor advisory of the latter problem, the licensee had initiated vendor-recommended design changes to the intercooler adapter housings which will add additional weldment and structural members to provide additional stiffness to the housings. Modifications to one intercooler adapter on the Division 2 diesel generator had been implemented and the remainder of the intercooler adapters were scheduled to be similarly modified during the first refueling outage. At that time, the intercooler adapters will be removed, permitting inspection of associated welds. Inspection of the guide vane attachment welds for degradation similar to that experienced at Grand Gulf can be accomplished at that time and is considered an open item (440/88015-01(DRP)).

5. Operational Safety Verification (71707)

The inspectors observed control room operations, reviewed applicable logs, and conducted discussions with control room operators during this inspection period. The inspectors verified the operability of selected emergency systems, reviewed tag-out records and verified tracking of Limiting Conditions for Operation associated with affected components. Tours of the intermediate, auxiliary, reactor, and turbine buildings were conducted to observe plant equipment conditions including potential fire hazards, fluid leaks, and excessive vibrations, and to verify that maintenance requests had been initiated for certain pieces of equipment in need of maintenance. The inspectors observed plant housekeeping/cleanliness conditions and verified implementation of radiation protection controls.

These reviews and observations were conducted to verify that facility operations were in conformance with the requirements established under Technical Specifications, 10 CFR, and administrative procedures.

In the course of these reviews, the inspectors determined that on October 7, 1988, the Control Room Emergency Recirculation System was rendered inoperable. The licensee failed to make the NRC notification required by 10 CFR 50.72. This matter is further discussed in Paragraph 6.c of this report.

No violations or deviations were identified.

6. Followup of Nonroutine Events at Operating Power Reactors (93702)

a. September 4, 1988 Offgas Hydrogen Burn and Charcoal Adsorber Ignition

On September 4, 1988, at approximately 9:25 A.M., while operating at 100 percent power, the "A" steam jet air ejector (SJAE) automatically isolated on sensed low flow. Over the next five minutes, operators placed the "B" SJAE in service. Upon placing the "B" SJAE in service, a slug of moisture was introduced into the offgas system. Offgas hydrogen concentration spiked to greater than 5 percent and then returned to normal (approximately 2 percent). Subsequently, two auxiliary operators dispatched to refill offgas system loopseals which had been lost during the system transient reported hearing a "pop" in the vicinity of the offgas system. At 10:35 A.M., operators noted that the temperatures of the 12A and 12B offgas charcoal adsorbers were gradually increasing and realigned the offgas system to bypass the 12A and 12B adsorber vessels. Based upon continued increases in adsorber temperatures, the licensee concluded that the 12A and 12B charcoal adsorbers had ignited and declared an unusual event at 12:10 P.M. All notifications required by the licensee's emergency plan were satisfactorily carried out within the following 15 minutes. By 12:30 P.M., reactor power was reduced and being maintained at 70 percent. At 2:32 P.M., the licensee established a nitrogen purge on the 12A and 12B adsorbers.

The licensee established continuous monitoring of adsorber temperatures and periodic analysis of offgas carbon dioxide and carbon monoxide concentrations. The 12A adsorber bottom thermocouple measured a peak temperature of 659.5 degrees F at 5:26 P.M., while the 12B adsorber bottom thermocouple measured a peak temperature of 582.2 degrees at 6:36 P.M. Subsequently, temperatures showed a declining trend and returned to their normal values by September 6, 1988.

On September 7, 1988, the licensee terminated the unusual event, based upon meeting criteria which indicated that combustion in the charcoal adsorbers had ceased. The criteria, which were discussed in advance with NRC Region III and NRR management personnel, were as follows: near normal and/or decreasing temperature readings on all offgas adsorber temperature instruments; and carbon monoxide concentration in the offgas effluent of less than 100 ppm for at least four consecutive hours. Criteria for reinstituting an Unusual Event if necessary, were also agreed upon by licensee and NRC management. The licensee returned to 100 percent power later the same day with the 12A and 12B charcoal adsorbers bypassed and maintaining a nitrogen purge on the adsorbers.

Detailed followup inspection of this event was conducted during a special team inspection conducted by the NRC Region III Division of Reactor Safety documented in NRC Inspection Report 440/88016(DRS).

b. September 16, 1988 Offgas Hydrogen Burn

On September 16, 1988, at approximately 9:20 P.M., while operating at 100 percent power, operators observed a decrease in generator electrical output, decreasing main condenser vacuum, and decreasing offgas system flows. In accordance with off-normal operating procedures, a power reduction was commenced. At 9:35 P.M., operators placed a second steam jet air ejector in service and increased nitrogen purge flow to the 12A and 12B charcoal adsorber vessels from 5 SCFM to 25 SCFM. Subsequently offgas system flow spiked high and licensee personnel reported a "boom" in the vicinity of the offgas building. Offgas hydrogen analyzer readings increased to greater than 5 percent. A second flow spike and percussion was observed. At 9:58 P.M., offgas effluent sample analysis indicated a carbon monoxide concentration of 2000 ppm. Based upon the foregoing evidence of a hydrogen burn in the offgas system, the licensee declared an unusual event at 10:02 P.M. and an orderly shutdown was commenced. The cause of the low offgas flow condition and degraded condenser vacuum was not immediately apparent. At 7:30 A.M. on September 17, 1988, the unusual event was terminated based upon offgas effluent carbon monoxide concentrations of less than 100 ppm. Unlike the September 4, 1988 offgas hydrogen burn event, charcoal adsorber temperatures remained normal throughout the event, indicating little, if any, charcoal ignition. The licensee achieved cold shutdown on September 18, 1988 and began a detailed investigation into the causes and circumstances surrounding the September 4 and 16, 1988, offgas hydrogen ignition events.

A special inspection team was dispatched from the NRC Region III office on September 19, 1988 to perform a thorough review of licensee actions to identify the causes of the offgas hydrogen ignition events and to assess the effectiveness of licensee corrective actions to prevent recurrence. The results of the special team inspection will be documented in NRC Inspection Report No. 440/88016(DRS).

Subsequent to this event and the special team inspection, the inspectors determined that the flow indicating switch which provided a "B" SJAE automatic isolation signal on low steam flow to protect against the accumulation of excessive concentrations of hydrogen in the offgas system routinely exhibited a positive offset of approximately 50 percent of full scale, even in the absence of any steam flow. Such a positive offset could conceivably defeat the low steam (dilution) flow isolation function. Resolution of this matter will be tracked as an open item (440/88015-02(DRP)).

c. October 7, 1988 Entry Into Technical Specification 3.0.3 As a Result of Inoperable Control Room Emergency Recirculation System

On October 7, 1988, at approximately 1:05 A.M., while operating at 100 percent power, an auxiliary operator touring the plant noticed that the "B" control complex water chiller was not running. While attempting to start the chiller, a control power fuse was found to be blown. The fuse was replaced and startup of the chiller was then reattempted. The new fuse blew and the operator locally observed sparks while attempting to start the chiller the second time.

Based upon the foregoing, the "B" control complex water chiller and the "B" train of the control room emergency recirculation system were declared inoperable at 1:30 P.M.. At the time of the occurrence, the "A" train of the control room emergency recirculation system was similarly rendered inoperable by virtue of the fact that the "A" control complex water chiller was out of service for planned maintenance. As a result, the licensee entered Technical Specification 3.0.3 at 1:30 P.M..

Troubleshooting and repair activities were commenced immediately and by 2:21 P.M., the cause of the "B" chiller inoperability had been identified and repaired. Degraded wire insulation had resulted in an electrical short to ground. The insulation had degraded from rubbing up against the vibrating enclosure in which the wire was located. Technical Specification 3.0.3 was thus exited at 2:21 A.M., prior to expiration of the one hour time limit for commencing a reactor shutdown.

In reviewing this event the following day, the inspectors determined that the licensee had failed to make a non-emergency NRC notification within 4 hours of declaring both trains of the control room emergency recirculation system inoperable as required by 10 CFR 50.72b(2)iii. The inspectors brought this matter to the licensee's attention and subsequently the licensee notified the NRC operations center via the ENS phone. Failure to report the loss of both control room emergency recirculation subsystems within the time limits specified in 10 CFR 50.72 is a violation (440/88015-03(DRP)).

7. Monthly Surveillance Observation (61726)

On August 29, 1988, the inspectors observed Technical Specifications required testing that was conducted in accordance with Surveillance Instruction (SVI)-C51-T0028-B, Revision 2, "Average Power Range Monitor (APRM) Flow Biased Signal Channel B Calibration for 1C51-K605B." On October 13, 1988, the inspectors observed Technical Specifications required testing that was conducted in accordance with Surveillance Instruction (SVI)-E12-T0161-B, Revision 1, "Emergency Core Cooling System/Low Pressure Core Injection (LPCI) A Discharge Pressure High Channel A Functional for 1E12-N565A."

For the above tests the inspectors verified that testing was performed in accordance with procedures, that test instrumentation was calibrated, that Limiting Conditions for Operation were met, that removal and restoration of the affected components were accomplished, that test results conformed with Technical Specifications and procedure requirements and were reviewed by personnel other than the individual directing the test, and that any deficiencies identified during the testing were properly reviewed and resolved by appropriate management personnel.

No violations or deviations were identified.

8. Open Inspection Items

Open inspection items are matters which have been discussed with the licensee, which will be reviewed further by the inspector, and which involve some action on the part of the NRC or licensee or both. Open inspection items disclosed during the inspection are discussed in Paragraphs 4. and 6.b..

9. Plant Status Meetings (30702)

On August 24, 1988 and September 30, 1988, NRC management met with CEI management at the Perry Site to discuss the current status of the plant, recent events and licensee initiatives to improve the quality of plant operating and maintenance activities. These meetings are being held on a periodic (initially monthly) basis.

10. Exit Interviews (30703)

The inspectors met with the licensee representatives denoted in Paragraph 1. throughout the inspection period and on October 18, 1988. The inspectors summarized the scope and results of the inspection and discussed the likely content of the inspection report. The licensee did not indicate that any of the information disclosed during the inspection could be considered proprietary in nature.