U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-423/85-35

Docket No. 50-423

License No. CPPR-113 Priority -- Category B

Licensee: Northeast Nuclear Energy Company
P. O. Box 270
Hartford, Connecticut 06101

Inspection at: Millstone Unit 3, Waterford, Connecticut

Inspection conducted: July 9 - August 12, 1985

Inspectors: Ebe McCabe, for 8/29/85
T. A. Rebelowski, Senior Resident Inspector Date

Ebe McCabe, for 8/29/85
F. Casella, Resident Inspector Date

K Ferlic 8/29/85
K. Ferlic, Project Engineer Date

Ebe McCabe, for 8/29/85
R. Summers, Project Engineer Date

Approved by: Ebe McCabe 8/29/85
E. C. McCabe, Chief, Reactor Projects Section 3B Date

Inspection Summary: Inspection 50-423/85-35 (July 9 - August 12, 1985)

Routine resident (125 hours) and region-based (86 hours) inspection including repairs to condenser air piping, auxiliary feed pump endurance runs, allegations, electrical terminations for diesel generators, a diesel fuel spill, steam generator wet layup, status of reactor water level monitoring, and status of the Safety Parameter Display System.

No unacceptable conditions were identified.

## DETAILS

1. Persons Contacted

Numerous members of Northeast Utilities and Stone and Webster (S&W) Corporation including engineers, technicians, craftsmen, and members of staff management were contacted.

2. Licensee Action on Previously Identified Item

(Closed) Inspector Follow Item (423/84-14-01 (a.7)): Effects of fire suppression system on vital equipment. The licensee has performed a review to determine the effect of fire suppression systems on vital equipment. A safety evaluation on the Emergency Diesel Generator (EDG) Enclosure Building concluded that one of the two diesels would be lost from actuation of the fire suppression system. The licensee has changed the automatic spray system in the EDG enclosure to a dry pipe (no pressure) manual system to minimize the potential for EDG loss. NRR is to perform a 10 CFR 50 Appendix R fire protection review prior to fuel loading and review the licensee's changes in the fire suppression systems. This inspector concern is therefore closed.

(Closed) Inspector Follow Item (423/84-02-07): Establish measures to protect equipment from grinding work. The licensee has established specific areas for grinding/welding activities. For these areas, protection for equipment and personnel have been provided. In addition, for grinding/welding activities that must be accomplished at field locations, the job supervisors have been instructed on accepted safe work practices to prevent equipment damage. Observations of field stations were made by inspector with no deficiencies noted.

3. Condenser Chloride Intrusion During Turbine Hot Functional Testing

On July 8, 1985, during turbine hot functional testing, a routine grab sample of the condensate pump discharge indicated an increase in the chloride concentration from approximately 60 parts-per-billion (ppb) to 29.4 parts-per-million (ppm). In-line instrumentation was unavailable to detect the chloride intrusion because the instrumentation had been isolated due to a constant alarmed condition (high conductivity) resulting from chemical additions (NaOH, Hydrazine). The chloride intrusion was limited to components associated with the condenser and condensate system. High chloride concentration was observed in the hotwell drain line (5,056 ppm), with associated components (condensate surge tank, auxiliary boiler B) tested and found to be approximately 20 ppm. The steam generators were isolated from the intrusion, with chloride concentrations remaining less than 30 ppb. In addition, there was no sea water intrusion into the turbines or moisture separation reheaters.

Licensee followup found that the flange connecting the air reduction pipe to the condenser tube sheet in the condenser D water box had separated from the tube sheet. Circulating water (sea water) then entered the secondary side of the condenser via the opening between the two. The silicon-bronze bolts which fastened the pipe flange to the tube sheet were severely corroded and the deteriorated heads had allowed the flange to separate from the tube sheet. Subsequent inspection of the other water boxes revealed all silicon-bronze bolts exhibited various stages of corrosion.

In addition, the inlet aluminum-bronze tube sheets exhibited various degrees of corrosion. Condensers C and D tube sheets exhibited the most extensive corrosion with pit cavity depths up to 450 mils. Other condenser components exhibited signs of corrosion. Although cathodic protection was installed in the water boxes, it was not operational in all cases. Even the inlet water boxes with cathodic protection exhibited an etched surface.

The licensee evaluated several potential corrective actions. As of August 2, the following corrective action was either in progress or scheduled to be implemented:

-- Cut the condenser tubes on the discharge side flush with the tube sheet to minimize turbulence and assure maintenance of a good epoxy seal.

-- Coat the air reduction pipes, inlet and outlet tube sheets, and inlet and outlet water boxes with a protective epoxy coating.

-- Spark test the epoxy coating to verify the minimum coating thickness.

-- Replace corroded stainless steel pipes with new stainless steel pipes.

-- Replace the silicon-bronze bolts with monel bolts.

-- Leak test the condenser tubes to verify integrity.

-- Review the cathodic protection system and perform several tests to establish the effectiveness of the system and make any appropriate changes.

-- Continue flushing contaminated lines and disassemble one sample valve subject to the chloride ingress to evaluate whether there is any corrosion damage to the valve.

-- Perform visual inspection of the Unit 3 water boxes and tube sheets on the same schedule as the operating Units 1 and 2.

The inspector made several observations in condenser water boxes D, E, and F for the above concerns. After review of the licensee's in-progress and scheduled corrective action, the inspector had no further questions.

While following the licensee's actions in response to the chloride intrusion, the inspector reviewed the current steam generator chemistry programs. The licensee has issued PORC approved procedures for Primary Chemistry Control (CP 3802A, Rev. 0), Secondary Chemistry Control (CP 3802B, Rev. 0), and Balance of Plant Chemistry Control (CP 3802C, Rev. 0). The licensee has been attempting to maintain the cold shutdown/wet layup chemistry requirement of the procedure since initial steam generator filling in early December 1984. Cold shutdown/wet layup requirements are established for nitrogen cover gas overpressure (3-5 psig), water level (above wet layup nozzles), pH (>9.8), hydrazine (>75 ppm), sodium (<1000 ppb), chloride (<1000 ppb), sulfate (<1000 ppb), and sampling (3/week). Actual sampling has been performed approximately 5-6 times a week since the steam generators were filled. Sodium and sulfate concentrations have not been followed due to the unavailability of analytical instruments. Chloride concentrations, hydrazine, and pH have been maintained within limits.

Operations does not yet have a procedure for maintaining the steam generators in a wet layup condition. Operations relies on chemistry personnel to inform them of the required conditions (i.e., nitrogen cover pressure and level). Formalized guidance for operators was being reviewed by the operations supervisor at the time of the inspection. This item will be reviewed during a subsequent inspection (IFI 50-423/85-35-01).

4. Bulletin Status Review

a. Licensee Review

The inspector discussed with a cognizant Nuclear Operations engineer the licensee's method for complying with IE Bulletins. The bulletins are dealt with in two phases: the first is determination of applicability and correction of the deficiency; the second is preventing recurrence.

Upon receipt of a Bulletin (or Notice or Circular), the Vice President of Nuclear Operations initiates a Nuclear Operations Assignment (NOA) to the Station Superintendent who then distributes it for action via a Controlled Routing (CR).

The CR provides a dedicated file for each Bulletin, Notice, or Circular, which contains all documentation generated during the engineering reviews as well as responses to NRC. If the deficiency is applicable, corrective action is either initiated immediately or picked up on the Commitment to Follow (CTF) program, depending upon priority.

Regardless of current applicability to the plants, the deficient items are added to the Nuclear Operations Deficient Items List (NODIL). The NODIL contains all deficient items, listed alphabetically by vendors, that have been identified by Bulletins, Notices, or Circulars. The NODIL is issued and maintained by Nuclear Operations, with controlled distribution, and is revised at least twice a year. It is referenced by Procurement Quality Assurance (PQA) personnel during each purchase order

(PO) review. Any NODIL item appearing on a PO is reported to Nuclear Operations for review. If the item is specifically prohibited, the purchase is not approved. In addition, there is currently an NOA in effect to review warehouse inventories against the NODIL to ensure deficient items are not in storage as replacement parts or spare components.

Based on the above administrative procedure, licensee actions on the following IE Bulletins were reviewed:

b. Bulletin Status

(1) Bulletin 76-02, "Relay Coil Failures - GE Type HFA, HGA, HKA, HMA Relays," described failures in listed relays identifiable by white nylon coils and manufactured prior to 1969. Stone and Webster Engineering Corporation (SWEC) performed an evaluation which showed that the relay types identified in the bulletin have not been used in safety related equipment unless they contained black lexan coil spools. The licensee responded to this bulletin by letter dated May 5, 1976. These 4 types of relays appear on the NODIL. Subsequent problems with lexan coil spools have been taken up by IE Bulletin 84-02. Bulletin 76-02 is closed.

(2) Bulletin 79-11, "Faulty Overcurrent Trip Device in Circuit Breakers for Engineered Safety Systems," described defects in the delay dashpots in Westinghouse Type DB-50 and DB-75 breakers. The licensee's investigation, documented in internal letter NEC-3306, stated that no circuit breakers of the type described are used at Millstone 3. Further, the NODIL contains these two breaker types to prevent future procurement. This bulletin is closed.

(3) Bulletin 79-24, "Frozen Lines," requested a review to determine that adequate protective measures have been taken to assure safety related process, instrument, and sampling lines do not freeze during extremely cold weather. The licensee has conducted this review and considers the design criteria established to be adequate. The design criteria are described in the licensee's response to the bulletin dated October 29, 1979. This bulletin is closed.

(4) Bulletin 79-27, "Loss of Non-Class 1E Instrumentation and Control Power System Bus During Operation," described a loss of power at Oconee. The licensee has reviewed this bulletin and concluded that Millstone 3 is designed to achieve cold shutdown without the use of any non-Class 1E power.

The station is designed in compliance with Reg Guides 1.139, "Guidance for Residual Heat Removal," and 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems." Equipment required to achieve a cold shutdown is redundant and is powered from redundant Class 1E buses. Loss of power on both Class 1E and non-Class 1E buses is annunciated on the main control board. This bulletin is closed.

(5) Bulletin 80-07, "BWR Jet Pump Assembly Failure" is not applicable to Millstone 3 and is closed.

(6) Bulletin 83-04, "Failure of the Undervoltage Trip Function of Reactor Trip Breakers," pertained to GE AK-2 type circuit breakers. Millstone 3 has Westinghouse DS-416 breakers. This bulletin is not applicable and is closed.

(7) Bulletin 83-06, "Nonconforming Materials Supplied by Tube-Line Corporation Facilities at Long Island City, New York; Houston, Texas; and Carol Stream, Illinois," requested a review to determine if Tube-Line supplied materials have been furnished to the facility, to identify systems on which the materials are used and to justify such use. SWEC determined that three vendors supplied Tube-Line materials to Millstone 3. The records further state that none of the materials have been used in safety related systems. Currently, the NODIL prohibits purchase of any Tube-Line materials. This bulletin is closed.

(8) Bulletin 83-08, "Electrical Circuit Breakers with an Undervoltage Trip Feature in Use in Safety Related Applications Other Than the Reactor Trip System," was issued to all facilities for action. It was issued to assure proper operation of circuit breakers with undervoltage trip attachment (UVTA's) being used in safety related applications other than as reactor trip breakers. By internal memo NNEC 3-2219, the licensee documented the results of a review which determined that the only breakers using UVTA's are the reactor trip breakers. This bulletin is closed.

5. Preoperational Test Witnessing

Throughout the inspection period the inspector witnessed various pre-operational tests performed by the licensee. The inspector verified testing was conducted in accordance with approved procedures and accurate, legible records of test results were documented. As applicable to the test witnessed, the following items were reviewed: latest revision of test procedures available, minimum crew requirements met (number of personnel and adequacy of knowledge of the testing requirements), proper prerequisites and associated plant systems in service, use of calibrated test equipment, adherence to criteria for test interruption and continuation, documentation of unusual events or discrepancies, adequate test performance and accurate data collection, proper tracking of temporarily installed modifications in accordance with administrative controls, meeting overall test acceptance criteria, preliminary test results consistent with observations, utilization of current drawings and manuals, and accurate incorporation of design changes or modifications in accordance with plant controls.

The following tests were witnessed:

-- Containment structural integrity test crack mapping.

-- Diesel generator 24 hour run testing.

-- Reactor coolant pump seal isolation valve testing.

-- Volume control tank vent valve testing and annunciator indication.

-- ESF charging isolation and control transfer to the auxiliary shutdown panel.

During the above testing an anomoly was identified between the plant computer display on the main control board and the main control board valve status lights. During testing of the reactor coolant pump seal isolation valves (CHS AV8141A, B, C, D and CHS MV8109A, B, C, D), the valve status displayed on the plant computer digital point display was displaying the valve open in green and the valve closed in red. This color coding is opposite that of the valve status lights on the main control board (i.e., green: closed or no flow condition; red: open or flow condition). Although operators are expected to rely on the control board indication for plant status, a quick viewing of the computer digital point display during a plant transient could mislead the operator as to actual valve status. The inspector was informed by licensee personnel that they would evaluate the color coding of the computer valve status display. This item will be followed (IFI 423/85-35-02).

6. Allegation Followup

a. RI-85-A-082

The resident inspector reviewed an allegation from an individual concerning various phases of installation of insulation on non-safety and safety related piping.

Concern 1: The alleger stated that he has a concern about the depth of insulation covering in-line valves in the house boiler (non-safety) used to supply steam for heating and preliminary testing of secondary systems.

Action Taken: The specifications allow gouging (reduction in thickness) of insulation to accommodate in-line valves. Examples of thickness requirements are specified for the safety related piping. The alleger's concern in the in-house boiler area is not safety related. An inspection of the area found no exposed valves that may constitute a safety hazard to personnel. The inspector has no further questions in this area.

Concern 2: The alleger stated that piping insulation metal material of a lighter gauge steel than required was used for insulation protective metal covers.

Action Taken: The inspector reviewed portions of Specifications 921 and 345 that delineate thickness requirements for insulation protective metal. Acceptable lower limits of thickness are 0.010 or 0.018 inches. An inspection of shop material was made of formed metal and sheets of aluminum

(non-containment) covering and stainless steel covering. Material was gauged by the inspector and found acceptable (0.021, 0.19, and 0.023"). No deficiencies in the field were observed.

Concern 3: Although not actually installing containment piping insulation, the alleger stated that he had talked with other insulators and learned that primary pipes were not being washed and dried with a solvent prior to installing insulation.

Action Taken: The inspector reviewed the licensee's method for the evaluation of chloride and fluoride surface contamination on stainless steel pipes prior to the installation of insulation (which is the basis for washing and drying of primary pipes before insulation installation). The inspection included a review of the applicable test procedures, logs and records, discussions with plant personnel, observations of sample preparation, surveying of a recently prepared stainless steel surface, and review of the procedures and authorization chain to clean and insulate stainless steel pipes.

Pipes are washed with demineralized water and/or an approved chemical solvent. The washed pipe need not be surveyed immediately after washing. In any case, the pipes are wrapped and taped in chloride/fluoride free materials to maintain cleanliness until survey results are analyzed and the pipe released for insulating. Surface surveys are partial and random, sampling a portion of the pipes cleaned. In the event pipes are unwrapped, re-survey is not required unless the pipes are uncovered for more than 24 hours. Upon receipt of favorable survey results, Quality Control releases the pipes for insulation. If pipes are found to have insulation removed and have not been evaluated for cleanliness, a Nonconformance and Disposition Report is issued by Quality Control identifying the suspect pipes. Documentation was clear, legible, and traceable. No deficiencies were noted.

When applicable, the dimensions of the surveys may be reduced to a surface area less than the recommended 2 ft. sq. area (e.g., surveys of small bore piping or areas in which small pieces of insulation have been removed as for inspecting a pipe weld). Based on the minimal amount of pipe covering that remains to be completed no additional changes were identified as necessary to meet specification requirements.

b. <u>RI-84-A-107</u>

The inspector reviewed aspects of allegation RI-84-A-107 pertaining to qualifications of personnel to torque support bolting. Previous followup on this allegation is documented in NRC Inspection Report 50-423/84-10. The alleger also identified, phoenetically, an individual who he thought was not qualified to install expansion anchor bolts. During this inspections the phoenetic identification was correlated to a specific person. The inspector reviewed the personnel training records for that individual and found that he had been qualified to install expansion anchor bolts on his first day of work onsite. The inspector had no further questions on this matter.

7. Plant Tours

During this report period, in addition to specific work observation, periodic plant tours were performed to observe general material conditions, housekeeping, and fire protection. Available ongoing work was periodically spot checked for use of proper procedures, calibrated/qualified tools, knowledgeable personnel, and overall good practices. The areas toured were: portions of the Turbine Building, Mainsteam Valve Building, Service Control Building, ESF Building, all levels and accessible compartments of the Auxiliary Building; and all levels of Containment including steam generator/reactor coolant pump bays, pressurizer space, accumulator spaces, incore instrument areas, and portions of the accessible compartment. No unacceptable conditions were identified.

8. Procedure Inadequacy

On July 19, 1985 while transferring diesel fuel from the A to B storage tank (Procedure 3346B, Revision 0, Diesel Fuel Oil), approximately ten (10) gallons of oil were spilled from the B storage tank vent. Plant Incident Report 105-85 was issued for licensee action.

The inspector reviewed procedure OP 3346B along with the actions of plant operators and the licensee's system for identifying corrective action. In accordance with the plant incident report procedure (ACP-QA-10.01), an investigator was assigned to investigate the event. NRC review of the event indicated that (1) the procedure in use failed to provide guidance to the operators as to when the transfer should be terminated, (2) a tank level sensor/indicator is available but it does not provide a high level annunciator alarm, (3) the computer output level of indication can display 0 to 100% tank capacity but no guidance is furnished to identify the fill level at which the tank overflows.

Inspector Observations

During the plant tour of the Diesel Storage Tank hold it was observed that, although the fill caps for storage tank A and B were locked by a padlock, the caps themselves could still be opened approximately one inch. The fill lines are readily accessible and could be a pathway for the addition of foreign materials to the fuel tanks. The above were discussed with licensee management. Correction of this item will be followed as an unresolved item, pending review of licensee action on Plant Incident Report 105-85 (UNR 423/85-35-03).

9. Reactor Water Level Monitoring System

With the installation of a Reactor Water Level Monitoring System (RWLMS), the licensee has addressed the TMI concern that reactor vessel water inventory information was not available to the operator after the incident.

General

The RWLMS consists of a Heated Junction Thermocouple System (HJTCS) with two thermocouple probe assemblies and signal processing equipment.

The probe assembly in each HJTCS channel consists of eight HJTC sensors, a separate tube, a seal plug, and electrical connectors. The sensors are physically independent and located at key level points between the reactor vessel head and the fuel alignment plate.

The HJTC sensor consists of a Chromel-Alumel thermocouple next to a heater (heated junction) and another Chromel-Alumel thermocouple positioned away from the heater (unheated or reference junction). In a fluid with relatively good heat transfer properties (water), the temperature difference between the adjacent thermocouples is very small. In a fluid with relatively poor heat transfer properties (steam), the temperature difference between the thermocouples becomes large. In the absence of water, the heated thermocouple temperature increases in relation to the unheated thermocouple, causing differential temperature to rise. At a predetermined setpoint, the system considers the sensor uncovered, changes the display level, and issues an alarm.

Observation

During a normal inspection tour of containment, observations were made of the shipment packaging of one reactor vessel level probe. The probe had experienced mechanical damage in that a mechanical protective fitting weld for the electrical cables was broken. Damage was tentatively identified as being due to the method of unscrewing the bullet-nosed lifting cap and its interference with internal cables. The probe was returned to the manufacturer for repair.

The inspector examined the instrument rack room, the minicomputer, and the operating station to observe testing of Reactor Vessel level circuitry. Annunciator windows have been established on the control board giving alarms for deviations of reactor vessel levels in the reactor head which then would require monitoring at the instrument rack. The testing of these annunciators will be performed during the normal Hot Functional Testing fill of the vessel.

10. Inspection and Rework on Diesel-Generator Terminations

The inspector reviewed licensee action on a Stone and Webster Engineering Company review of Colt Industries QA/QC activities pertaining to crimped terminations in Diesel Generator panels 3EGS*PNL-1B, 3EGS*PNL-B, and 3EGS*TBEG-1B (E and DCR FE-06020). Action included inspection of lugs against the Millstone panel vendor crimp criteria for identification and replacement of unacceptable crimps along with a review of the consistency between panel and drawing identification numbers. Observation was performed of the inspection and rejection of crimps, qualification of tools, and replacement of crimps. Documentation was being maintained in accordance with the engineering design change. The inspector observed no unacceptable conditions.

11. Auxiliary Feed Pump - Electric Driven

References: a. Test Procedure T3322 - 1 M01
b. Probablistic Safety Study, Part 1, Volume 1
c. TMI Action Plan Item E.1.2, Item E.1.1

General

The licensee is conducting the initial preoperational testing of the motor driven Auxiliary Feed Pump. The mechanical checkout includes a two hour performance run, bearing temperature stabilization, vibration readings, coupling alignments, and the forty-eight (48) hour endurance run to Standard Review Plan 10.4.9.

In addition, the feed pumps are involved in a number of probablistic event scenarios (Reference b) which denote the absence of the ability to supply auxiliary feedwater to mitigate events, as a prominent contributor to these events (Reference c).

Observations

The regional and resident inspectors have previously witnessed auxiliary feedwater wear ring modification and bearing changes (due to overheating). Cleanliness of oil systems appeared to be one contributor to the bearing problems.

The licensee performed a forty-eight (48) hour run on each pump from July 24 thru 26, 1985. The inspector's observations included, prior to operation, the coupling alignments; monitoring of motor running current (37-40 amps/phase), which were satisfactory; and test gauges on suction and discharge lines, which were within their calibration cycles. The test pressures were matched with control room board indications and found in agreement.

Suction strainer removal was not done upon completion of the test. This item will be checked prior to completion of the Hot Functional Testing (IFI 85-35-06).

Findings

No deficiencies in completed testing were identified. The following two concerns were identified.

a. One area of logic testing was not documented in that with motors in the remote pull-to-lock position, the ability to regain local control and start motor was not demonstrated. The licensee demonstrated this feature to the inspector for one motor. Results were satisfactory. The licensee added this item to the procedure for both motors.

b. The feedwater pump bearings are drained to a pump sump by means of tygon tubing with adjustable hose clamps securing the hoses. This item is a possible pathway for introduction of contaminates, could be easily severed, could be subject to heat softening. This item is to be reviewed by the licensee (IFI 423/85-35-04).

12. Safety Parameter Display System (SPDS) - NRR Audit

References:
a. NUREG-0737, Clarification of TMI Action Plan Requirements - Item I.D.2
b. NUREG-0737, Supplement I, Section 4
c. Northeast Utilities - Docket Nos. A02959, A04508, A04615, and A04752

General

On July 29-30, 1985, the inspector participated in NRR's review of the SPDS machine-operator interface and of the basic knowledge necessary to respond to action levels indicated by CRT displays.

a. Presentations

Licensee presentations included:

-- Nuclear Engineering Section involvement,
-- Display Design Methodology,
-- SPDS Use Philosophy,
-- Display Descriptions,
-- Project Documentation - Process Computer Engineering,
-- Functional Specification,
-- Independent Verification of Functional Specification,
-- SPDS Verification and Validation,
-- SPDS Man-In-The-Loop Validation,
-- MP-3 Signal Validation Process,
-- SPDS Availability, and
-- Plant Process/SPDS.

Detail reviews of specifications and the SPDS process were conducted by members of the NRR Audit Team.

b. Observations and Ease of SPDS Use - Practical Demonstration

The licensee has developed a video instruction tape that develops an accident scenario that calls on the SPDS displays to appear on CRTs that would indicate an accident. The six elements displayed on the SPDS are subcriticality, core cooling, heat sink status, RCS integrity, containment status, and RCS inventory. The program runs for approximately 45 minutes and allows operators to note changing conditions in the core, containment, and RCS.

c. Site Status

The SPDS has not been completed at the site. The onsite process computer program for SPDS is not complete. Training for operators has not been accomplished. Schedules for training and the training course outline were requested by the senior resident inspector. This item will be reviewed at a subsquent inspection (IFI 423/85-35-05).

13. NTOL Status Review

The inspector reviewed documents related to issuance of an operating license. The licensee's construction and testing status and the NRC's inspection program status were reviewed. Outstanding items, including commitments to NRR and the status of licensee actions in regard to previous NRC inspection findings, IE Bulletins, Circulars, and Information Notices were reviewed and prioritized. The licensee also provided an assessment of QA/QC activities during the construction phase of the project for review. The inspector had no further questions at this time; however, this area will be reviewed by resident and region-based inspectors during the preoperational test phase.

14. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable or not. An unresolved item identified during this inspection is discussed in Detail 8.

15. Exit Meeting

At periodic intervals during the course of this inspection and on August 8, 1985, meetings were held with senior plant management to discuss the scope and findings of this inspection. No proprietary information was identified as being in the inspection coverage.