U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-423/85-74
Docket No. 50-423
License No. NPF-44 Category B

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

Facility Name: Millstone Nuclear Power Station, Unit 3

Inspection At: Waterford, Connecticut

Inspection Conducted: November 19, 1985-January 6, 1986

Inspectors: T. A. Rebelowski, Senior Resident Inspector, Millstone 3
F. A. Casella, Resident Inspector, Millstone 3
J. T. Shedlosky, Senior Resident Inspector, Millstone 1/2
H. H. Nicholas
R. J. Summers, Project Engineer

Approved by: Ebe McCabe 3/6/86
E. C. McCabe, Chief, Reactor Projects Section 3B Date

Inspection Summary: Inspectio 50-423/85-74, 11/19/85-1/6/86

Areas Inspected: Routine onsite regular and backshift inspection by the Senior Resident and Resident inspectors (381 hours) and an NRC Region I inspector and contractor employee (36 hours). Areas inspected included: review of licensee action on previous findings, review of NUREG 0737 action items, witnessing of system and component testing, observation of the initial core load; inspection of surveillance, maintenance, construction and plant physical protection activities; and operational safety verification reviews.

Results: Four violations were identified. The first, (Appendix A, Item A, and Detail 19), dealt with maintenance and construction personnel failing to maintain cleanliness control on safety-related systems which were opened up for modification or inspection. The second, (Appendix A, Item B, and Detail 18) was a failure by operations personnel to follow a written procedure to restore a safety-related system to its normal configuration after an infrequent evolution. The third and fourth violations were isolated instances related to physical security, (Appendix B and Detail 13).

Notable license strengths were found in the continued professionalism and high morale of the startup organization and in the conservative and safe approach taken by management and the operations staff during the initial fuel load.

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DETAILS

1. Persons Contacted

J. Ferland, President
J. Opeka, Senior Vice President, Nuclear Engineering and Operation
R. Werner, Vice President, Generation and Construction Engineering
W. Romberg, Station Superintendent
J. Crockett, Unit 3 Superintendent
F. Rothen, Construction Superintendent

The inspector also contacted other licensee employees during the inspection, including members of the Operations, Radiation Protection, Chemistry, Instrument and Control, Maintenance, Reactor Engineering, Security and Training Departments.

2. Facility Activities Summary

The NRC issued operating license NPF-44 to Millstone Point Unit 3 in the afternoon on November 25. The first fuel assembly was loaded in the reactor core approximately 26 hours later; detailed technical specification review and Mode 6 surveillance testing were completed during the interim. Fuel load proceeded for 6 days, with the final of 193 assemblies loaded by approximately 11PM on December 2.

There were delays in the fuel load process, principly due to difficulties with the spent fuel pool crane and Sigma refueling machine. The spent fuel pool crane load limit device repeatedly exhibited an intermittent open circuit that could not be identified. As corrective action, the licensee plans to relocate the device to a more vibration free area of the bridge and install a repeater for operator interface. The Sigma refueling machine had problems with indexing, vertical positioning, drive motor degradation, and disc brake flutter. At times, it was necessary to position and lower fuel assemblies manually.

One fuel assembly (A35) was bowed approximately 1/4" over its twelve foot length, preventing the adjacent assembly (B49) from landing on the core plate. The licensee modified the loading sequence to "box in" the B49 location. Subsequently, the assembly was properly landed.

After the initial fuel load, numerous maintenance and construction activities were undertaken and pushed transition to Mode 4 and commencement of post-core-hot-functional-testing beyond the period of this report. As part of the stress reconciliation program, the four Containment Recirculation System (RSS) heat exchanger tube side nozzles were strengthened and the Quench Spray System Chemical Addition Tank (QSS-CAT) supports were upgraded.

Other work included modifications to Supplementary Leak Collection and Release System (SLCRS) damper control circuits, motor control centers and fan control breakers. A number of SLCRS boundary door seals had to be improved before the SLCRS draw down test was completed satisfactorily. Both Emergency Diesel

Generators (EDGs) underwent complete fuel system inspections and had fuel return lines replaced. A check valve in the Primary Grade Water System (PGS) supply to the Charging and Volume Control System (CVCS) was replaced.

Component Cooling Water (CCP) system containment isolation check valves were lapped in and upstream flanges were installed to allow Local Leak Rate Testing (LLRT). In addition, CCP containment isolation valve motor operators were replaced to assure timely closure against full system flow. These changes were undertaken to safeguard against a loss of coolant accident outside containment in the event of a tube failure in the reactor coolant pump thermal barrier heat exchangers.

A Reactor Coolant System pressure boundary leak occurred when the B loop RTD bypass manifold vent line socket weld failed due to cyclical fatigue. The socket was rewelded and the line configuration was modified to prevent recurrence. Further vibration testing of other small bore piping subject to Reactor Coolant Pump (RCP) induced oscillations led to additional work in upgrading the supports to the B RCP seal injection line and reanalysis of the seal injection line supports on the remaining 3 pumps.

At the completion of this report period, the plant remained in Mode 5.

3. Licensee Action on Previous Inspection Findings

A. (Closed) Unresolved Item (85-62-04), Calibration of CO2 Concentration Measurement Instruments

The licensee was able to mix a 40% concentration by volume of CO_2 in air. The 2 Truure instruments were able to measure this known concentration within the +/- 2% accuracy provided by the manufacturer. In each of the 8 areas tested, the Truure instruments were used to take data. These data show conformance with acceptance criteria. This item is closed.

B. (Closed) Unresolved Item (85-54-01), Seismic Category 1 Acceptability of Control Room Pressurization Air Storage Tanks

The inspector did not find the installation acceptable due to construction deficiencies, lack of QC acceptance criteria, and incomplete QC inspection.

A QC inspection was performed shortly after the above discrepancies were made known. The inspector reviewed Stone and Webster QA Inspection Report 5A00154, originated 9/23/85, which performed the QC inspections on the air storage tank supports required by E&DCR TR-02761. The specific construction deficiencies from this unresolved item were included in the list of findings attached to the QC inspection report. The findings were specific and well documented and resulted in Nonconformance and Disposition Reports 15390, 15530, and 15381. The inspector reviewed portions of these N&Ds for rework and disposition.

There was a questionable "accept as-is" disposition. One QC finding was that some fastener nuts did not have sufficient thread engagement because the required 1/4" of bolt was not visible. This was disposed "accept as-is". The inspection questioned the disposition. The cognizant engineer stated that threaded rods were the fasteners in question, and that they had square ends. The 1/4" visibility factor was based on ensuring full thread engagement when chamfered bolt ends were in use. The engineer stated that all fasteners checked had full engagement of the threads on the rods and nuts. The inspector had no further questions on this item.

All discrepancies listed as unresolved in the Seismic Category 1 inspection have been completed. This item is closed.

C. <u>(Closed) Unresolved Item (423/85-06-01) Approved Procedures forwarded to NRC for Review 60 days prior to Scheduled Test</u>

The licensee has scheduled draft procedures of the start-up and power ascension program to meet the 60 day criteria. During this inspection, the NRC received draft start-up and refueling procedures within the above time frame. This item is closed as being no longer applicable.

D. <u>(Closed) Unresolved Item (423/84-07-01) Review Plant Design Change Request Control Document to Improve Management Control</u>

This item is addressed in ACP-QA-3.10, Preparation, Review and Disposition of Plant Design Change Requests (PDCRs) (NEO). An extensive revision of Nuclear Engineering and Operation Procedure NEO 3.03 based on review of PDCRs at the Haddam Neck Plant. This item is closed.

E. <u>(Closed) Unresolved Item (423/85-02-03) Review of Service Water Transient Test Results</u>

The licensee has analysed and tested various piping configurations to determine the induced pressure transients caused by the partial draining of service water, with an accompanying air gap, between restarts of service water pumps. In-line check valve closing propagates pressure waves throughout the service water system. The tests pinpointed the service water system supply to a Ventilation Water Chiller as a contributor to these pressure transients. The licensee modified piping to reduce the air gap formation. Additional testing verified that pressure transients had been reduced to an acceptable operating condition. This item is closed.

F. <u>(Closed) Security Outstanding Items (85-64-01)</u>

The licensee completed closure of security plan commitments prior to issuance of the operating license.

The resident inspector performed verification inspections in a number of areas including:

-- Verifying barriers and vital area closures for thirty-eight areas including gates, fences, drains, inlet and outlet ventilation areas, diesel enclosures, and witnessing the use of increased patrols and additional CCTV monitoring.

-- Protected Area Barriers in areas of drainage were inspected and patrols were observed.

-- Intake structure and hatch protection concerns were resolved.

-- Lighted areas including patrols, trailer lights, and areas that did not previously meet criteria.

-- Detection aids and compensatory measures.

-- Upgraded training of supervision and security force.

All of the above areas were found acceptable. This item is closed (85-64-01).

G. <u>(Closed) Violation (IV) (85-423/85-12-03) Flooding of Engineered Safety Features Building (ESFB)</u>

As documented in the licensee's June 26, 1985 response to the May 28, 1985 report, the licensee's actions included an investigation of the cause. Correction actions were reviewed by the Plant Operation Review Committee (PORC) and the Joint Test Group (JTG). Included in this review was the Flush Program activities in progress and their controls, test activities released and currently in progress, and maintenance activities in progress. The results of licensee review indicated a failure to follow the Automated Work Order tagging requirements. Startup Engineers were briefed on requirements to follow administrative program controls. The above actions did not prevent additional flooding incidents which are discussed in NRC Reports 85-16 and 85-23. A program review, performed in response to the above incidents, resulted in a tightening of flush controls. Those corrective actions did result in a clarification of flush program interties.

The resident inspector witnessed area cleanup, removal of all wetted insulation, examination of RHR bearings and retest of systems that were involved in flooding (electrical circuit retest, valve packing replacement, etc.) No deficiencies were observed. This item is closed.

H. (Closed) Violation IV (85-423/85-16-01) Flooding of a portion of Control Building

The licensee investigation indicated that tagging for the Automated Work Order (AWO) was adequately performed and in place. The licensee failed to verify that the isolated service water system was drained prior to removal of a valve for maintenance. Shift Supervisors, Supervising Control Operators, and Maintenance department personnel were briefed on the incident. Corrective actions included an examination of flooded area electrical cabinets and battery rooms. Shutdown panels in the 4'6" level appeared to have soaked control wire. No wiring required replacement. The resident inspectors witnessed the cleanup. Two successful tests of the remote shutdown system were performed. The administrative controls on verification of maintenance prerequisites did not have to be modified because the event was due to failure to comply with established practices. Maintenance practices will be reviewed further under the NRC operational inspection program. This item is closed.

I. (Closed) Violation IV (85-423/85-23-01), Fire Protection Water Flushing With Improper Lineup

The licensee failed to tag a boundary valve during flushing. Licensee investigation concluded that personnel error and a weakness in administrative Startup Manual procedures were root causes. The licensee revised the Startup Manual (Rev 4) to ensure that interrupted test conditions, which was a contributing factor in this incident, will be reviewed. Additional controls on documenting system changes are in place. No further flooding incidents have occurred. This item is closed.

4. Reported Significant Construction Deficiency (CDR)

(Closed) CDR 84-00-08, Main Steam Line Break Outside Containment. The potential for a high energy line break outside containment to affect environmentally qualified equipment was identified by the licensee on June 5, 1984, with the Main Steam Valve Building being the primary concern. As a member of the Westinghouse Owner's Group, the licensee reviewed the basis for the postulated breaks, temperature and pressure profile development criteria, resulting temperature and pressure profiles, and consequent results. To satisfy NPF-44 licensee condition 2.C(3), the licensee presented to NRR an analysis concluding that there is adequate protection to satisfy concerns related to this issue. The NRC found this acceptable as documented in Paragraph 3.11 of Supplement 5 to the Safety Evaluation Report (NUREG-1031). This CDR is therefore closed.

5. Review of Worker Concerns

A. Allegation RI-85-A-47

An FQC inspector stated he had knowledge of inspection aspects that other FQC inspectors did not have but was directed by his management to cease instructing other workers. An on-site individual was identified as being knowledgeable. That onsite individual was contacted by the NRC Region I specialist inspector who followed up on this alleger's concerns. Also, the NRC Region I Projects Section Chief for Millstone 3 contacted that on-site individual by telephone. That individual stated that he considered the alleger to be knowledgeable and that, to his knowledge, the individual had been instructed not to instruct other inspectors in the areas involved because the individual's experience at another site may not be directly correlatable to Millstone 3. The NRC concluded that it is the licensee's prerogative to designate instructors and that no construction inadequacy had been identified in this matter. Adequacy of construction and construction FQC has been routinely assessed during NRC inspections. The overall conclusion, as documented in NRC Region I Report 50-423/85-67, is that there has been acceptable construction, quality assurance, and quality control. Therefore, no additional follow-up on this allegation has been prescribed.

B. Allegation RI-85-A-105

This allegation pertains to QC inspection activities.

The first concern was that management failed to take corrective actions for concerns about drawing control. The alleger identified drawings as not being of the proper revision. Drawing control had been a previous NRC concern for which the licensee had implemented corrective actions. It was determined by the inspector that these actions had been implemented and that the drawings identified by the alleger were of the proper revision. Two other concerns, pertaining to improper storage of drawings and non-traceability of inspection records of cable tray supports in the control building, had been identified by the alleger to the licensee's quality concern organization. The inspector determined that the licensee was taking action in accordance with their allegation program. The licensee's preliminary findings for the above concerns were that: (i) the drawing revisions were proper; (ii) improper drawing storage was not substantiated; and, (iii) the inspection records were, in fact, traceable. As noted in Inspection Report 50-423/85-67, the as-built Millstone 3 design has been found, through multiple inspections, to conform to requirements. The inspector had no further questions on drawing control.

The second concern pertains to discrepancies on support anchor bolts which had been previously inspected and found acceptable. The inspector reviewed the alleger's inspection reports and determined that a total of 11 anchor bolts were identified as having less than the specified embedment on 4 supports located in the turbine building on the outside wall of the control building. As alleged, these anchor bolts had been previously accepted. However, the check which identified the discrepant condition was a scheduled final inspection.

The nonconforming conditions identified by the alleger were dispositioned in accordance with the licensee's procedures. The support was classified as QA Cat 1 because it was attached to the Seismic Category 1 control building wall; however, the cable trays supported were on the turbine building side and are neither seismically qualified nor safety-related. The licensee had identified the cause of the erroneous earlier inspection as inspector error in interpreting an E&DCR written for the supports. The E&DCR addressed whether rebar could be cut in order to get the required embedment depth for the anchor bolts. The licensee reviewed other QC inspection reports by the individual who made the faulty inspection to take corrective actions where the same error was made. In this case, defects identified were not numerous and were properly dispositioned, affected cable trays were not safety-related, and the lack of embedment of these anchor bolts does not adversely affect the safety-related wall. The inspector had no further questions on this item.

The third concern was that the alleger was terminated after identifying these concerns to the NRC. The inspector determined that the alleger was laid off in a planned reduction in force based on seniority. The alleger had been employed for less than a year and this layoff included QC inspectors with greater seniority than the alleger. The inspector had no further questions on this concern.

6. <u>Licensee Actions taken as a result of TMI Action Plan Requirements Specified in NUREG-0737</u>

The NRR staff reviewed the licensee's submittals associated with these items and discussed the results of this review in the Safety Evaluation Reports (SER) related to the operation of Millstone Nuclear Power Station Unit 3. During this inspection, the licensee's actions as described in the SER were verified to have been taken.

A. <u>I.A.1.1.1 & 3 (85-TM-01) Shift Technical Advisors (STA) (Closed).</u>

The licensee had numerous meetings with NRR on the use of STAs at Millstone 3 and has committed to provide "Shift Advisors" for shifts which do not have the prescribed hot operating experience. On obtaining required operating experience, an individual who is SRO licensed on Millstone 3 and who holds a bachelors degree in engineering, engineering technology or a physical science may serve in a dual role of SRO/STA. Otherwise, a dedicated STA who meets the criteria of NUREG 0737 is to be on each operational shift. The inspectors verified that STA and Shift Advisor staffing meets Technical Specification and Operating License requirements, and have reviewed and witnessed training of Shift Advisors. No deficiencies were identified. This item is closed.

B. A.1.1.3.2A & B (85-TM-03) Shift Manning (Closed).

The shift manning as described in the Technical Specifications and the Shift Operating Experience have been subject to License NP-44 condition 14, which required certain hot operating experience. The licensee's present four shifts, expanding to six with NRR approval of Shift Advisors based on training program and simulator training, have been monitored by NRR, NRC Region I and the resident inspectors. Shift complement meets Technical Specification and operating license requirements. This item is closed.

C. II.G.1 (85-TM-19) Emergency Power for Pressurizer Equipment (Closed).

The licensee's plan to satisfy General Design Criteria 10, 14, 15, 17, and 20 of Appendix A to 10 CFR 50 for the event of loss of offsite power required implementation of circuit design to assure reliable power supplies. Millstone 3 has one PORV receiving power from 125V vital dc, with its associated block valve receiving power from the same 480V ac emergency bus. The second PORV and block valve has a similar arrangement but is powered from the opposite electrical division. The review of drawings EE-1AQ-B and EE-1BS-8 verifies this arrangement.

Specific Configurations:

1) Orange Train

Valve	Power
3RCS SOV 455A	(PORV)---------BATT 1 (0) 3 PYS-22F
3RCS MV 8000A	(Block Valve)--32(0)-2R (RHF)

2) Purple Train

Valve	Power
3RCS SOV 456	(PORV)---------Batt 2 (P) 3PYS-23F
3RCS MV 8000B	(Block Valve)--32(P)-2W (RHF)

The licensee addressed this item to the NRC in letter B11511, April 11, 1985 (Counsil to Youngblood). This item is closed.

D. III A.1.1 (85-TM-22) Emergency Preparedness (Closed)

The NRR review of the Technical Support Center discusses the location and facilities. The resident inspectors have witnessed the use of the center during a Unit 3 walk-thru of a simulated nuclear incident. The support center has three separate work stations, one for each unit, with design drawings and CRT displays of plant system data. NRC facilities for 2 to 3 persons are provided. Communications to the Control Room and Emergency Operating Facility were acceptable. This item is closed.

E. III A.1.2 (85-TM-23) Upgrade of Emergency Support Facilities (Closed)

The licensee has a combined site upgraded Emergency Plan and a Corporate Nuclear Incident Plan. NRC Emergency Preparedness Inspections (Reports 50-423/85-39; 85-66) have found the licensee's emergency planning facilities and controls to be acceptable under current standards. This item is closed.

F. III.D.3.3 (85-TM-24) Improved Inplant Iodine Instrumentation under Accident Conditions (Closed)

The licensee has placed airborne monitors in areas that sample the air from the reactor containment, the engineering safety features building, the control room, and locations in the reactor plant heating and ventilation air streams. In addition, portable instrumentation is available for use where workers may be exposed to a high radiation field. Procedure AOP 3573, Radiation Monitor Alarm Response, and Procedure EPIP 4225, Containment Air Post Accident Sampling, were reviewed with the finding that the licensee has met the commitments documented in SER 12.3.4.2. This item is closed.

7. Licensee Action on Safety Evaluation Report (SER) Open Items

The licensee has presented to the Office of Nuclear Reactor Regulation (NRR) a number of physical and procedural items to resolve NRR concerns. This inspection verified the following items.

A. (Closed) SER Section 7.4.2.4, Remote Shutdown Operation (85-SE-05) The licensee has developed two tests for remote shutdown outside the control room. TP 5018 was performed during the preoperational test program. TP 8029 will be performed during power ascension testing. In addition, EOP 3505, "Shutdown Outside Control Room" and EOP 3504, "Cooldown Outside Control Room" have been written and are used during shutdown testing. Regional inspectors witnessed TP 5018 testing and found the results acceptable. TP 8029 testing acceptability is an integral part of the NRC field inspection program. Based on the licensee having provided the committed features and an acceptable testing program which will receive separate NRC review, this item is closed (85-SE-05).

B. (Closed) SER Section 8.3.2.2 D.C. Monitoring (85-SE-06) The generic requirements in IEEE STD 308-1974 state that the DC system (batteries, distribution systems and chargers) shall be monitored to the extent that it is shown to be ready to perform its intended function. The NRC staff has accepted some remote monitoring. NRC Inspection Report 423/85-54 Page 15 thru 18 discusses the areas of concern. In addition, Technical Specification 3/4.8.2, D.C. Sources, denotes requirements for these systems. The alarms for main board annunciators are found on Main Boards 1-MB1D, 1-MB1E, 8-MB8A, and 8-MB8C. Operating procedures address responses to alarms that indicate possible D.C. anomalies. The inspector determined that TS 3/4.8.2 was being met. This item is closed (85-SE-06).

C. (Closed) SER Section 8.3.2.3, Compliance to Generic Letter 81-04 (85-SE-07) The license has developed procedures (3500 series) that address the loss of power. The inspector reviewed the licensee's program for classroom training, Lesson Plan 3504, and Simulator Instructors Guide #CLC(S-11). The licensee has completed the commitments documented in the Safety Evaluation Report. This item is therefore closed (85-SE-07).

D. (Open) SER Section 9.3.2.2, PASS System (85-SE-08) The regional inspection of PASS systems that verify the 11 criteria of NUREG 0737 Item II.B.3 is scheduled when enough radionuclides are present in the coolant to perform a meaningful test. The resident inspectors witnessed the licensee's ability to line up systems and obtain a representative PASS sample. Mechanical leaks were observed and corrected prior to conclusion of this test. EPIP 4226, Core Damage Estimate Procedure, addresses the techniques and EPIP 4224, Post Accident Sampling System, address the methods of sampling. These include such parameters as preplanning, prior to sampling, of stay times, routes, respiratory protection, and dosimetry. These items will be inspected at a subsequent inspection. This item remains open (85-SE-08).

E. (Closed) SER Section 10.4.9, Emergency Procedure for Backup Water Supply (85-SE-12) The licensee has described the various sources of water to maintain the Steam Generator as a heat sink in operational procedure OP 3322. Paragraph 7.6 describes shifting of auxiliary feedwater pump suction to the service water system. Length of stay of the introduced contaminents is addressed in the procedure. The inspector has witnessed the licensee's successful placement of the spool pieces to tie the feedwater suction to the service water system. The licensee has satisfactorily addressed the SER commitment. This item is closed (85-SE-12).

F. (Closed) SER Section 15.9.11, Report on Outages of ECCS (85-SE-13) The licensee has issued NOP-2.12, Participation in Industry Data Programs, such as Nuclear Plant Reliability Data System (NPRDS) and the Generating Availability Data System (GADS) covering system and component failures, outages and power reductions, licensee event reports, etc. The licensee has implemented the above systems, satisfying their commitment on this item. This item is therefore closed (85-SE-13).

G. (Closed) SER Section 10.4.2, Mechanical Vacuum Pump Exhaust Monitoring (85-SE-11) The licensee's commitment to continuous sampling for iodines and particulates in the mechanical vacuum pump has been met. Procedure OP 3329, Condenser Air Removal, has been revised to require chemistry to set up a monitoring system and to remove the monitoring system upon conclusion of vacuum pump operations. This item is closed (85-SE-11).

8. Contact with Waterford First Selectman

The Senior Resident Inspector met with the Waterford First Selectman in the Selectman's office on January 2, 1986. The meeting was held to maintain communication between town officials and the NRC resident office. Discussions centered around the status of Millstone Unit 3 and included the non-radioactive steam discharges with high noise levels that would be associated with hot functional testing and the power ascension programs. The inspector asked the Selectman to feel free to call on the resident staff if any questions should arise.

9. Witnessing of Initial Fuel Load

A. Scope

The resident inspectors, assisted by regional office based inspectors and resident inspectors from Millstone 1 and 2 and Haddam Neck, observed the fuel load, which began on November 26 and ended on December 2.

Test procedure 3-INT-4000, the overall controlling document for the fuel load process, was approved by the Plant Operations Review Committee (PORC) on November 8, 1985. This procedure referenced various plant operating and surveillance procedures for completion of routine requirements as well as five appendices for completion of specific fuel load tasks. The Appendices were: 4002, "Operational Alignment-Nuclear Instrumentation Systems;" 4003, "Core Load Instruments and Neutron Source Requirements;" 4004, "Inverse Count Rate Ration Monitoring;" 4005, "Initial Core Loading;" and 4006, "Core Map." This entire 4000 series of procedures had been reviewed and found acceptable by Region I specialists.

B. Verification of Activities

The following activities were verified to be in accordance with the facility Technical Specifications and the 4000 series procedures during the six day fuel loading sequence.

1. Communications were established via headsets between the spent fuel crane operator, the Sigma refueling machine operator, a control room operator and the reactor engineer's staff.

2. All fuel movements were tracked on a status board in the control room. The board indicated fuel pit locations, core locations, and fuel-assembly-in-transit locations (i.e., spent fuel crane, fuel transfer tube, Sigma refueling machine).

3. The reactor engineer directed fuel assembly movements with the concurrence of the Shift Supervisor. Watchbills were in force. Even distribution of experienced watchstanders over the 3 shift rotation was verified. Qualified Plant Equipment Operators (PEOs) manipulated the spent fuel bridge and transfer tube machinery. Licensed

reactor operators handled the Sigma refueling machine. A Senior Control Operator [Senior Reactor Operator (SRO) License] was on the refueling machine and in charge of refueling floor (Containment) operations whenever core alterations were in progress.

4. Source range neutron level was continuously monitored by audible count rate in the control room and on the refueling floor.

5. Containment integrity was set and maintained during core alterations in accordance with Technical Specification Section 3/4.9. The equipment access hatch was closed, the transfer canal was flooded above the transfer tube elevation, and at least one door to the personnel access hatch was shut at all times. The containment purge and exhaust dampers were open when the exhaust radiation monitor was in service and closed when the monitor was not in service in accordance with Technical Specification 3.9.9.

6. Primary sources (inserts A23PS1 and B23PS2) were inserted in the correct assemblies (C30 and C04 respectively) which were loaded first and in close proximity to the two source range excore detectors to put the source range channels on scale. Neutron dose rates were monitored by Health Physics personnel during the movements of these two assemblies.

7. Class II Housekeeping zones were established and maintained in the spent fuel pool area and on the refueling floor around and above the reactor cavity.

8. The reactor coolant system was flooded to above the loop nozzles with water having a boron concentration of about 2050 ppm. The 1.6% shutdown margin was determined to be 1850 ppm boron. Sampling and analysis of boron concentration was performed at different locations in the system periodically. Both Residual Heat Removal (RHR) trains were operable with at least 1 train always in service. A boron flow path with two sources of borated water (Refueling Water Storage Tank and Boron Addition Tank) was always available.

9. In addition to the permanent source range detectors, 3 portable submersible "dunkers" were installed in the core and were shifted around in accordance with the Appendix 4005 loading sequence to monitor the changing core geometry. The portable detectors output went to signal processing equipment, counter/scalers, and strip chart recorders on the refueling floor.

10. The reactor engineering staff evaluated the response of each loaded assembly by extrapolation of the inverse count rate ratio plot before allowing the loading of the next assembly.

11. Excore nuclear instrumentation surveillances were performed at required frequencies. The "high flux at shutdown" alarms were in normal for both source range channels for the entire time.

12. Fuel load test procedure 4000 results reviews are documented in Details 11B and 11C.

C. Problems encountered during fuel load

1. Sigma Refueling Machine. The licensee encountered numerous problems with the refueling crane. While moving to pick up the second fuel assembly, the operators shut the machine down because of a loud noise whose source could not be determined. Maintenance determined the cause to be excessive movement of the trolley disc brake mechanism causing a flutter when the crane was moved at high speed. The brakes were adjusted and the operators were cautioned not to travel at maximum speed.

While the sixth assembly was being lowered into the core, an operator noticed a loose 1/4 X 20 X 1 1/4" bolt resting on a cluster plate in the crane insertion tube. The assembly was lowered. Fueling was halted for inspection of the Sigma machine. The absence of 2 bolts with nuts and washers from a junction box above the insertion tube was revealed. Engineering inspected the reactor vessel with an underwater camera but did not find the missing fasteners. It was noted that the fasteners could have been disengaged prior to fuel load. The licensee concluded that there was no safety concern from the loss of the missing material. The missing fasteners were replaced. Fuel load was restarted. The resident inspector witnessed the discovery of the bolt and examined the area of the missing fasteners.

There were delays due to difficulties with the positioning and load limit switches. On one occasion, while lowering a fuel assembly into the core, the hoist height limit switch went out of adjustment, causing the lower limit to actuate before the assembly was unloaded. The Sigma logic concluded there was nothing for the assembly to land on. After visual verification of proper positioning, the assembly was successfully manually lowered, unloaded, and ungrappled.

Finally, the trolley drive motor seriously degraded during the fuel load. Prior to core mapping, the drive motor was replaced.

2. Spent fuel pool crane. The other major cause of delay in the initial fuel load was spent fuel crane breakdowns. The major contributor to these was an intermittent open in the load limit control circuit that could not be identified through numerous troubleshooting attempts. To correct this problem, the licensee intends to relocate the load limit circuitry to a more vibration free area of the bridge and install a repeater for operator interface.

3. Bowed fuel assembly. On November 30, the operators were unable to land assembly B49 at core location E-4 due to interference with assembly A35 (location F-4). It was determined that A35 was bowed approximately 1/4 inch over its 12 foot length, which was enough to keep B49 from contacting its guide pins on the lower core plate. The licensee called a Plant Operations Review Committee (PORC) meeting at 0300 on 30 November (meeting 3-85-399), with Westinghouse representation, to approve Change 5 to INT-4000, Appendix 4005, which modified the loading sequence. Assembly B49 was temporarily landed at the northern perimeter of the core while the sequence continued and "boxed in" its E-4 location. On December 1, B49 was loaded successfully.

D. Conclusion

Fuel load went well. Actions were conservative and deliberate. Operator training and performance were appropriate.

10. Preoperational Test Results Review

A. The inspector reviewed five completed preoperational test procedure packages ready for licensee review, evaluation, and approval of test results as listed below.

(1) Preoperational Test Results Evaluation Reviewed

--- T3322-P Auxiliary Feedwater System
--- T3313-AP Hydrogen Recombiner and Building Ventilation Systems
--- T3314-EA Service Building HVAC System
--- T3314-FP Control Building HVAC System
--- T3410-BP Reactor Vessel Level

No discrepancies were noted and test deficiencies were documented.

B. The inspector reviewed twenty-three completed preoperational test folders for deficiencies, including tracking and resolution, as listed below.

(1) Completed Preoperational Test Package Test Deficiencies Reviewed

--- T3314-DO ESF Building HVAC
--- T3315-BA Main Steam Valve Building HVAC
--- T3316-AP001 Main Steam
--- T3322-P Auxiliary Feedwater
--- T3330-CP Reactor Plant Chilled Water
--- T3330-EP Safety Injection Pump Cooling
--- T3307-AP003 Safety Injection Accummulator Test
--- T3307-AP001 Low Pressure Safety Injection
--- T3306-P Containment Recirc Spray
--- T3313-FP Containment Vacuum
--- 3-INT-2001 APP R03 Data Reduction Grid Alignment

--- T3308-P001 High Pressure Safety Injection Flow Balance
--- T3340-BA Water Treating
--- T3341-BP Fire Protection-Halon
--- T3344-BA050 MCC-32-1k
--- 3-INT-2001 App J03 RCS Leakage
--- 3-INT-2001 Aoo J09 Rx Eng-NSS Data Sheet
--- 3-INT-2001 R02 Reasonability Check and Time Coverage
--- T3308-P002 HP Safety Injection
--- T3314-FP Control Building HVAC
--- T3321-AP Feedwater and Recirc
--- 3-INT-2002 ILRT and SIT Test
--- 3-INT-2001 App J12 RP Prerated Water Inventory

No inadequacies were noted in the review of these preoperational test deficiencies. The inspector verified the tracking and proper resolution of the closed test deficiencies. The remaining open test deficiencies have been adequately re-tied to later modes and inspections.

11. Startup Test Program Review

A. The inspector reviewed three approved and one draft startup test procedures for technical and administrative adequacy as listed below.

(1) Startup Test Procedures Reviewed

--- 3-INT-5000 App 5033, RCS Loop Stop Valve and Pump Interlocks
--- 3-INT-5000 App 5017, Precritical RCS Flow Coasdown Measurement
--- 3-INT-6000, Initial Criticality-Controlling Procedure
--- 3-INT-6000 App 6001, Inverse Count Rate Ratio

No discrepancies were noted in the review of these procedures.

B. The inspector reviewed five completed startup test procedure packages ready for licensee review, evaluation and approval of test results as listed below.

(1) Startup Test Results Evaluation Reviewed

--- 3-INT-4000, Initial Fuel Load-Controlling Procedure
--- 3-INT-4000 App 4003, Core Load Instruments and Neutron Source Requirements
--- 3-INT-4000 App 4004, Inverse Count Rate Ratio Monitoring
--- 3-INT-4000 App 4005, Initial Core Loading
--- 3-INT 4000 App 4006, Core Map

No discrepancies were noted. Test deficiencies were documented.

C. The inspector reviewed five completed startup test folders for test deficiencies, their tracking and resolutions, as listed below.

(1) Completed Startup Test Package Test Deficiencies Reviewed

--- 3-INT-4000, Initial Fuel Load-Controlling Procedure
--- 3-INT-4000 App 4003, Core Load Init. and Neutron Source Req.
--- 3-INT-4000 App 4004, Inverse Count Rate Ratio Monitoring
--- 3-INT-4000 App 4005, Initial Core Loading
--- 3-INT-4000 App 4006, Core Map

No inadequacies were noted. The inspector verified the tracking and proper resolution of the closed test deficiencies.

12. Test Observation

A. The inspectors witnessed portions of various Post Core Hot Functional Tests. These included INT-5000 Appendix 5008, "Rod Drop Testing," both cold no flow and full flow; INT-5000 Appendix 5033, "RCS Loop Stop Valve Interlocks and Pump Interlocks;" INT-5000 Appendix 5016, "Loose Parts Monitoring System;" and INT-5000 Appendix 5031, "Chemical and Volume Control System."

Test performance was monitored for conformance to test procedures, operation of equipment in accordance with plant operating procedures, and good engineering practices. There were no discrepancies noted.

B. Secondary Leak Collection and Recovery System (SLCRS) Phase II testing was closely monitored by the inspectors. The licensee had numerous problems getting SLCRS boundary door seals to function properly and having fans and dampers operate as designed. As a result, a number of tests failed to meet the acceptance criteria of 0.25" water gauge negative pressure throughout the containment enclosure and contiguous buildings within 50 seconds. Extensive repair and upgrading of equipment was accomplished between test failures.

Prior to the final successful performance of T3314IP, the inspector walked down all 43 SLCRS boundary doors with the cognizant Startup Engineer. No discrepancies were noted. The final test met the acceptance criteria.

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14. Reactor Coolant System Pressure Boundary Leak

A leak on the B reactor coolant loop RTD bypass manifold high point vent line was discovered by a Plant Equipment Operator (PEO) on rounds on 1/3/86. The leak was very slow (a few drops per hour) and was detected by presence of boric acid crystal growth on components below the manifold. The leak was located at the toe of the socket weld that attached a short run of 3/4" 316 SS pipe to a sockolet attached to the manifold. The other end of the vertical pipe has a manual isolation valve topped by a 1500 psi rated blank flange. There are no supports connected to this pipe run.

Initial inspection revealed a clean crack at the toe of the weld against the pipe. It appeared to be fatigue induced. Licensee vibration analysis indicated that this section of unsupported pipe was in resonance with the B Reactor Coolant Pump (RCP). That was considered to be the cause of the cyclical fatigue crack. The remaining 3 bypass manifold vent lines were vibration tested and found not to be in resonance with their respective RCPs. (The B loop vent line was longer than the other three.)

Corrective Action: Automated Work Order (AWO) M3-85-00213 was issued to perform repairs. The weld was ground out, with the crack preserved for further failure analysis, and the 3/4" pipe was shortened by approximately 5 1/2" to eliminate resonance. A new weld was completed in accordance with ASME Section XI based repair program EM 31100 and Station Procedure DWP-101 "Fillet Welds on SS plate, pipe and fittings using GTAW process," Revision 3, dated 2/12/85. The new weld was nondestructively examined by visual and dye penetrant testing with satisfactory results. Quality control performed inspections at all requisite hold points. Hydrostatic pressure testing was not required because piping of less that 1" minimal is exempted by ASME Section XI 1WA-4000 Section 4400(b)5. A visual leak test was performed at 400 psi and the AWO remains open pending a system leak test at 2250 psi. Vibration measurements were made on the new configuration with satisfactory results. The inspector will review the final operational hydrostatic test results during a subsequent inspection (IFI 85-74-06).

15. Reactor Coolant Pump P-1 Snubber Support

Northeast Utilites informed the NRC staff of an error made in the calculations associated with the design of the load path for the Reactor Coolant Pump P-1 Snubber support. The P-1 snubber is one of several large supports associated with each of the four Reactor Coolant Pumps. The NRC was first informed of this design error during a September 12, 1984 meeting between the licensee and NRR personnel. The licensee provided additional details of the problem in a letter to the NRC (Serial A04477) dated December 7, 1984. In addressing the issue of not providing a Construction Deficiency Report, the licensee's stated position was that the error was discovered during the design verification process and, since the design was not final, the error was not considered reportable. The inspector reviewed this event. The error was found to have occurred in a manual calculation which was made to determine loads at the support during 1974. It was discovered during re-analysis of these loads in

1981. At that time, new analytical techniques were available. After correction of the original error, these new loads were found to exceed the design loading of the concrete embedments associated with two of the four reactor coolant loops.

The error did not appear to be generic to other primary support members because it occurred during the manual transformation of load coordinates and resulted from the change of an arithmetic sign for one component of displacement. The coordinate transformation was needed to allow loads generated by the cold leg rupture restraint gap analysis to be used in the dynamic analysis. The error was found using advanced computer techniques which had not been available earlier. These would have disclosed similar errors in other supports.

Although the new, correct loads were within the capacity of the P-1 snubber, they exceeded the design capacity of the concrete embedments used in Reactor Coolant Loops 1 and 2. The concrete corbels used to transmit the loads from support P-1 embedment to the containment structure are slightly different in loops 1 and 2, and were found to be limiting.

The safety issue was resolved on October 10, 1984 when calculations were completed to show that, in their as-built condition, the loop 1 and 2 embedments were capable of carrying the correctly calculated loads. This was due to the margin which existed in the original installation beyond the design minimum of the corbel. The calculations are stated in the "Qualification of the P-1 Snubber Support Bracket", [Calculation Number 12179-NS(B)-158-20C dated October 10, 1984.]

The licensee supported his position that the installed restraint design was not final until 1984 by identifying the number of load calculations made for the P-1 restraint. From its initial conceptual design, which was made on January 5, 1974, through final stress reconciliation on October 17, 1984, there have been nineteen (19) load calculations made for the P-1 support. Sixteen (16) of these were made before finding the error on November 6, 1981. The significance of the problem was stated by the licensee in its letter to the NRC (Serial B11295) dated September 12, 1984.

The issue of whether the licensee should have submitted a Construction Deficiency Report was discussed with the licensee at the Region I offices on January 9, 1986. Resolution of this consideration will be documented incident to that meeting.

16. <u>Observation of Stress Reconciliation Program Repairs</u>

The inspector observed work in progress during the reinforcement of the nozzle pads on the tube sides of the 4 Containment Recirculation System (RSS) heat exchangers. Reinforcement was determined to be necessary as a result of the stress reconciliation program; the as-built loading of these nozzles exceeded design bases. Work was being performed in accordance with E&DCR T-5-07940 under AWO M3-85-38023. The inspector observed gusset fitup, control of welding rod and welding in progress. There were no discrepancies noted.

17. Routine Periodic Inspections

A. Numerous plant tours were conducted during this inspection period to observe activities in progress and verify compliance with administrative requirements. Systems and equipment in areas toured were observed for fluid leaks and abnormal vibrations. Snubbers and restraints were observed for proper conditions. Plant housekeeping conditions were observed for cleanliness controls and fire hazard prevention.

During a tour of the auxiliary building on 12/20/85 at noon, the inspector noted 2 valves with significant leakage to atmosphere on the 4'8" level. One was the Charging and Volume Control System (CVCS) letdown containment isolation valve (CHS*CV8152), the other was primary grade water (PGS) relief valve (PGS*RV77). It appeared that both valves had been leaking for a significant period. There was a large area of crystalline boric acid on the floor under CHS*CV8152 and a deep and widespread puddle under the drain funnel leading away from the tailpiece on PGS*RV77. There were no existing trouble reports on either of these valves. With the unit in operation, leakage from CHS*CV8152 would be potentially contaminated. The inspector will follow this as IFI 85-74-05.

During the period of this report, the inspector made numerous tours of safety-related spaces to inspect work in progress. He noted that the auxiliary and service buildings exhibited numerous instances of grafitti and poor cleanup practices. Observations of painting of buildings and systems were included in these tours. In all cases, the inspector observed proper surface preparation prior to application of coatings.

B. Shift logs and operating records were reviewed periodically to determine the status of the plant as well as changes in operational conditions since the last log review. In addition, the following verifications were made; selected Technical Specification limits were met, operating logs and surveillance sheets were complete and log reviews were conducted by operating staff, and operating and night orders did not conflict with technical specification requirements. No deficiencies were noted.

18. Review of Plant Events

A. Diesel Fuel Oil Spill

Approximately 40 gallons of diesel fuel oil were spilled onto concrete and asphalt surfaces northeast of the emergency diesel generator building at 2PM on 11/21. The source of the oil was the flame arrester on B Fuel Oil Storage Tank as the tank was overflowed due to an incorrect valve lineup. The midshift on 11/21 performed a transfer from A to B storage tanks by pumping through a cross connect line and overflowing the B day tank back to the B storage tank. This line-up was in accordance with OP 3346B Rev 0, Change 2, "Diesel Fuel Oil System." Apparently, step 7.5.5 of that procedure was not completed because the system was not restored to a normal line-up. Early in the afternoon of that same day, Diesel Generator A was being run with Day Tank level control in Automatic. However, level in the day tank was constantly lowering, even though the transfer pump was running. A Plant Equipment Operator (PEO) was dispatched to check the position of the transfer pump discharge valve. He found it closed and opened it. No further checks were made. The operators were confident that valve lineups performed on 11/19 (for B) and 11/20 (for A) diesels were current. "A" day tank level was restored. Simultaneously, however, day tank B was overflowing to storage tank B, which eventually overflowed into the yard (Violation 85-74-02).

Immediate corrective action was witnessed by the inspector. The storm drains were unaffected-sandbags were quickly placed to minimize oil spread. Fire hoses were broken out and the spill was covered with sand and swept up. The storage tank vault was pumped out to waste drains. The inspector was concerned that the flame arrestor on the storage tank overflow might have been compromised. The licensee showed that the arrestor was a stacked metal disc type and that saturation was not a problem.

19. Cleanliness Control

A. During a tour of the Emergency Diesel Generator (EDG) enclosures for purposes of observation of inspection work being performed on EDG-A, the inspector noted the absence of material and personnel accountability while critical portions of the EDG were opened up. Specifically, 4 crankcase inspection covers (2 on each side) had been removed to permit access to the crank shaft, journal bearings and cylinder liners. The catwalks, below the open ports, contained loose tools and debris. Further, access to the internal area of the engine was uncontrolled. Maintenance and contractor personnel were not taking the precautions required to prevent items such as pencils and rulers from falling out of pockets while leaning over the crank or looking up into the cylinder liner.

B. During a routine tour of the lower level of the containment building on November 25, 1985, the inspector noted that some of the Containment Structure Sump (Engineered Safety Features Sump) protective deck plates were out of position. Further observation revealed that the sump was completely opened and that there was an unobstructed pathway directly to the Containment Recirculation System (RSS) pump suctions. Various tools and fasteners were randomly placed on the remaining horizontal plates above the sump and on the sump floor. There were no personnel present at the sump at the time of this observation. Licensee management was immediately informed of this situation.

Work was being performed by construction forces under Construction Work Permit (CWP) M3-85-36279, authorized for performance on 11/13/85, to relocate the sump level instrument 3RSS-LE49 as per Engineering and Design Change Request (E&DCR) TC-05339. That post-turnover work order did not specify any housekeeping/material accountability requirements. Stone and Webster Specification 2200.000-914, "Mechanical Equipment Erection," Section 3.4, "Housekeeping Requirements," states that housekeeping during construction shall be in accordance with ANSI N45.2.3, "Housekeeping During the Construction Phase of Nuclear Power Plants." ANSI N45.2.3 specified material and personnel accountability for Housekeeping Zone III systems. Station Administrative Control Procedure ACP-QA-2.02C, "Work Orders" lists the Containment Recirculation System (RSS) as a Zone III system.

Immediate corrective action was taken by the licensee to cover the RSS pump suctions with plywood taped to the sump floor. Subsequently, the suction lines were inspected using a boroscope with no foreign materials detected.

Later in the report period, the inspector again observed that work was being performed in the RSS Sump. New grating clamps were being installed on the RSS Sump Vortex Grating under PMMS Work Order M3-85-39595. This order was written to maintain cleanliness per Specification M914, (Stone and Webster Specification 2200.999-914), with specific instructions that the sump be lined with poly with plywood covers over the suction pipes. However, worker conformance to these administrative stipulations did not appear to be thorough. The material accountability list, posted inside the protective tent over the sump access, had a list of tools with no indication of when they were checked in or out. Further, there were rags and knee padding in the sump that were not included on the list. The pump suction penetrations were covered with plywood. The inspector expressed further concern to licensee management that cleanliness requirements were not being followed carefully enough.

The failure to maintain material and personnel accountability for both the EDG sump and the RSS sump are a violation (85-74-01).

20. Exit Meeting

At periodic intervals during the course of this inspection, 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. At no time during the inspection was written material provided to the licensee by the inspector.

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[X] New Item

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OUTSTANDING ITEMS FILE
SINGLE DOCKET ENTRY FORM

Docket # 50-423

CASELLA — Originator

REBARWSKI — Reviewing Supervisor Name

Item Number	Type	Module #	Area	Resp	Action Due Date (MM-DD-YY)	Updt/Clsout Rpt #	Date O/M/Clsd (MM-DD-YY)
85-74-03	NC5	71707	SPP	Z/U	04-06-86	- -	01-06-86

Originator: SHEDLOSKY

Modifier/Closer:

Descriptive Title: FAILURE TO MAINTAIN ESCORTS FOR WORKERS REQUIRING ESCORTS

Item Number	Type	Module #	Area	Resp	Action Due Date (MM-DD-YY)	Updt/Clsout Rpt #	Date O/M/Clsd (MM-DD-YY)
85-74-04	NC4	71707	SPP	Z/U	04-06-86	- -	01-06-86

Originator: CASELLA

Modifier/Closer:

Descriptive Title: FAILURE TO LOCK UNATTENDED VEHICLES IN PROTECTED AREA

Item Number	Type	Module #	Area	Resp	Action Due Date (MM-DD-YY)	Updt/Clsout Rpt #	Date O/M/Clsd (MM-DD-YY)
85-74-01	~~[illegible]~~ NC4	71707	PSC	Z/U	04-06-86	- -	01-06-86

Originator: CASELLA

Modifier/Closer:

Descriptive Title: FAILURE TO MAINTAIN CLEANLINESS CONTROL WITH SAFETY SYSTEMS OPEN TO MAINTENANCE

Item Number	Type	Module #	Area	Resp	Action Due Date (MM-DD-YY)	Updt/Clsout Rpt #	Date O/M/Clsd (MM-DD-YY)
85-74-02	NC4	71707	PSC	Z/U	04-06-86	- -	01-06-86

Originator: CASELLA

Modifier/Closer:

Descriptive Title: FAILURE TO FOLLOW OPERATING PROCEDURE IN RESTORING VALVE LINE-UP

Transaction Type

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☒ New Item

☐ Modify

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OUTSTANDING ITEMS FILE
SINGLE DOCKET ENTRY FORM

Docket # 50-423

CASELLA — Originator

REBELOWSKI — Reviewing Supervisor Name

Item Number	Type	Module #	Area	Resp	Action Due Date (MM-DD-YY)	Updt/Clsout Rpt #	Date O/M/Clsd (MM-DD-YY)
85-24-05	IFI	71707	MNT	Z/U	04-06-86	- -	01-06-86

Originator: CASELLA

Modifier/Closer:

Descriptive Title: VALVES IN LOWER LEVEL AUX BLDG EXHIBIT EXCESSIVE LEAKAGE WITH NO TROUBLE REPORTS SUBMITTED

Item Number	Type	Module #	Area	Resp	Action Due Date (MM-DD-YY)	Updt/Clsout Rpt #	Date O/M/Clsd (MM-DD-YY)
85-24-06	IFI	71707		/	- -	- -	- -

Originator: CASELLA

Modifier/Closer:

Descriptive Title: REVIEW OPS HYDRO FOR ACCEPTABILITY OF LOOP B RTD ~~BYP~~ VENT BYPASS LEAK REPAIR

Item Number	Type	Module #	Area	Resp	Action Due Date (MM-DD-YY)	Updt/Clsout Rpt #	Date O/M/Clsd (MM-DD-YY)
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Originator:

Modifier/Closer:

Descriptive Title:

Item Number	Type	Module #	Area	Resp	Action Due Date (MM-DD-YY)	Updt/Clsout Rpt #	Date O/M/Clsd (MM-DD-YY)
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Originator:

Modifier/Closer:

Descriptive Title:

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Transaction Type

- [x] New Item
- [x] Modify
- [ ] Delete

Docket # 50-423

Originator: CASELLA

Reviewing Supervisor Name: REBELOWSKI

Item Number	Type	Module #	Area	Resp	Action Due Date (MM-DD-YY)	Updt/Clsout Rpt #	Date O/M/Clsd (MM DD YY)
85-62-01	UNR			/	- -	85-74-C	01-06-86
85-51-01	UNR			/	- -	85-74-C	01-06-86
85-06-01	UNR			/	- -	85-74-C	01-06-86
85-SE-05	OTH			/	- -	85-74-C	01-06-86

Originator | Modifier/Closer

Descriptive Title

Transaction Type

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OUTSTANDING ITEMS FILE
SINGLE DOCKET ENTRY FORM

Docket # 50-423

CASELLA — Originator

Reblowski — Reviewing Supervisor Name

Item Number	Type	Module #	Area	Resp	Action Due Date (MM-DD-YY)	Updt/Clsout Rpt #	Date O/M/Clsd (MM-DD-YY)
85-SE-06				/	- -	85-74-C	01-06-86

Originator | Modifier/Closer

Descriptive Title

Item Number	Type	Module #	Area	Resp	Action Due Date (MM-DD-YY)	Updt/Clsout Rpt #	Date O/M/Clsd (MM-DD-YY)
85-SE-07	UNR			/	- -	85-74-C	01-06-86

Originator | Modifier/Closer

Descriptive Title

Item Number	Type	Module #	Area	Resp	Action Due Date (MM-DD-YY)	Updt/Clsout Rpt #	Date O/M/Clsd (MM-DD-YY)
~~85-SE-08~~				~~/~~	- -	~~85-74-C~~	~~01-06-86~~

Originator | Modifier/Closer

Descriptive Title

Item Number	Type	Module #	Area	Resp	Action Due Date (MM-DD-YY)	Updt/Clsout Rpt #	Date O/M/Clsd (MM-DD-YY)
85-SE-12	UNR			/	- -	85-74-C	01-06-86

Originator | Modifier/Closer

Descriptive Title

Transaction Type

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Docket # 50-423

Originator: CASELLA

Reviewing Supervisor Name: REBELOWSKI

Item Number	Type	Module #	Area	Resp	Action Due Date (MM-DD-YY)	Updt/Clsout Rpt #	Date O/M/Clsd (MM DD YY)	Originator	Modifier/Closer	Descriptive Title
84-07-01	UNR			/	- -	85-74-	01-06-86		REBELOWSKI	
85-02-03	UNR			/	- -	85-74-	01-06-86		REBELOWSKI	
85-64-01	UNR			/	- -	85-74-	01-06-86		REBELOWSKI	
85-12-03	NC4			/	- -	85-74-	01-06-86		REBELOWSKI	

Transaction Type

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- [x] Modify
- [ ] Delete

OUTSTANDING ITEMS FILE
SINGLE DOCKET ENTRY FORM

Docket # 50-423

Originator: CASELLA

Reviewing Supervisor Name: REBELOWSKI

Item Number	Type	Module #	Area	Resp	Action Due Date (MM-DD-YY)	Updt/Clsout Rpt #	Date O/M/Clsd (MM DD YY)	Originator	Modifier/Closer	Descriptive Title
85-16-01	NC4			/	- -	85-79-C	01-06-86		REBELOWSKI	
85-23-01	NC4			/	- -	85-79-C	01-06-86		REBELOWSKI	
85-TM-01	OWR			/	- -	85-79-C	01-06-86		REBELOWSKI	
85-TM-03	OWR			/	- -	85-79-C	01-06-86		REBELOWSKI	

Transaction Type

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OUTSTANDING ITEMS FILE
SINGLE DOCKET ENTRY FORM

Docket # 50-423

CASELLA — Originator

RESELOWSKI — Reviewing Supervisor Name

Item Number	Type	Module #	Area	Resp	Action Due Date (MM-DD-YY)	Updt/Clsout Rpt #	Date O/M/Clsd (MM-DD-YY)
85-TM-19	UNR			/	- -	85-74-C	01-06-86
85-TM-22	UNR			/	- -	85-74-C	01-06-86
85-TM-23	UNR			/	- -	85-74-C	01-06-86
85-TM-24	UNR			/	- -	85-74-C	01-06-86

Originator | Modifier/Closer

Descriptive Title

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CASELLA — Originator

REBELOWSKI — Reviewing Supervisor Name

Item Number	Type	Module #	Area	Resp	Action Due Date (MM-DD-YY)	Updt/Clsout Rpt #	Date O/M/Clsd (MM-DD-YY)
85-SE-13	UNR			V	- -	85-74-XX	01-06-86

Originator: | Modifier/Closer:

Descriptive Title:

Item Number	Type	Module #	Area	Resp	Action Due Date (MM-DD-YY)	Updt/Clsout Rpt #	Date O/M/Clsd (MM-DD-YY)
85-SE-11	UNR			V	- -	85-74-	01-06-86

Originator: | Modifier/Closer:

Descriptive Title:

Item Number	Type	Module #	Area	Resp	Action Due Date (MM-DD-YY)	Updt/Clsout Rpt #	Date O/M/Clsd (MM-DD-YY)
85-SE-08	UNR	7	SME	P/1	09-23-85	85-74-	01-06-86

Originator: CASELLA | Modifier/~~Closer~~: REBELOWSKI

Descriptive Title: REQUIRES REGIONAL HP INSPECTION FOR FINAL CLOSEOUT

Item Number	Type	Module #	Area	Resp	Action Due Date (MM-DD-YY)	Updt/Clsout Rpt #	Date O/M/Clsd (MM-DD-YY)
84-00-08	CDR			V	- -	85-74-	01-06-86

Originator: | Modifier/Closer:

Descriptive Title: