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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 7, 1996

MEMORANDUM TO: William Dean, Regional Coordinator
Office of the Executive Director
for Operations

FROM: Phillip F. McKee, Director
Northeast Utilities Project Directorate
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

SUBJECT: BRIEFING PACKAGE FOR VISIT BY NORTHEAST UTILITIES

Attached is the briefing package on the status of Millstone Nuclear Power Station, Units 1, 2, and 3, and the Haddam Neck Plant as background for the visit by Northeast Utilities on June 11, 1996.

Docket Nos. 50-213, 50-245
50-336, 50-423

Attachment: As Stated

cc w/atts: W. Russell
R. Zimmerman
S. Varga
J. Andersen

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I/9

SCHEDULE OF COMMISSION/SENIOR STAFF VISITS

11:00 a.m.	Chairman Shirley Jackson
1:00 p.m.	Commissioner Greta Dicus
2:00 p.m.	Commissioner Kenneth Rogers

BRIEFING PACKAGE FOR VISIT BY NORTHEAST UTILITIES

Visiting Officials:	Bernard M. Fox	President & Chief Executive Officer Northeast Utilities
	Robert E. Busch	President, Energy Resource Group * Chief Financial Officer Northeast Utilities
	Ted Feigenbaum	Executive Vice President & Chief Nuclear Officer Northeast Utilities
	Terry Harpster	Director, Nuclear Licensing Services Northeast Utilities

Date of Visit: June 11, 1996

Plants: Millstone Nuclear Power Station, Units 1, 2, and 3
Haddam Neck Plant

Note: Northeast Utilities (NU) is a holding company whose subsidiaries include: the Northeast Nuclear Energy Company (NNECO), the licensee for Millstone Nuclear Power Station, Units 1, 2, and 3; the Connecticut Yankee Atomic Power Company (CYAPCO), the licensee for Haddam Neck; and the North Atlantic Energy Service Corporation (NAESCO), licensee for Seabrook Station, Unit No. 1.

MILLSTONE NUCLEAR POWER STATION, UNITS 1, 2, AND 3

Millstone Station Senior Management Meeting Briefing Package (Enclosure 1)

Special Inspection Team - Millstone Quick Look Report (Enclosure 2)

HADDAM NECK PLANT

Haddam Neck Senior Management Meeting Briefing Package (Enclosure 3)

Special Inspection Team - Haddam Neck Quick Look Report (Enclosure 4)

SUPPORTING DOCUMENTATION

Common Northeast Utilities Configuration Management Plan (Enclosure 5)

Northeast Utilities Letter Dated May 31, 1996 - Challenges (Enclosure 6)

Biographical Data (Enclosure 7)

Organization charts (Enclosure 8)

Millstone Station Senior Management Meeting Briefing Package

Attachment 1

MILLSTONE UNITS 1, 2 AND 3

SALP Period: Because of the continuing high level of problems, particularly at Units 1 and 2, and because Millstone Station was added to the Watch List, a SALP report was not issued for the assessment period of July 10, 1994 through November 4, 1995 (the SALP board for this period was held on December 17, 1995).

SALP Ratings for the SALP report issued August 26, 1994:

Operations	*2,3,2
Maintenance	*2,3,2
Engineering	2
Plant Support	2

* Units 1, 2 and 3, respectively; Engineering and Plant Support ratings are composite evaluations.

I. PERFORMANCE OVERVIEW

This is the tenth SMM since June 1991 at which a Millstone facility has been discussed. In early 1992, in response to an overall decline in performance, the licensee implemented a Performance Enhancement Program (PEP) as a long-term effort to ensure the effective use of resources and implement the recommendations of four internal performance assessment task forces. The PEP did not have specific indicators to measure effectiveness and it had only limited impact on improving operational performance, particularly on Unit 2, and more recently on Unit 1. In October 1994, the PEP was absorbed into the utility's overall "business plan." Following the January 1995 SMM, it was determined that a visit by NRC senior managers to the Board of Trustees was an appropriate means of communicating the NRC's concern over Millstone Unit 2's continued poor performance. This meeting took place on March 17, 1995.

The NRC remains concerned about continued evidence of unresolved safety concerns at Millstone. Two 10 CFR 2.206 petitions filed in the last year to express safety concerns to the NRC, especially a petition related to core off loading practices at Millstone Unit 1, indicate that the licensee is still struggling to appropriately address safety concerns raised by its personnel. Site management has not been fully effective in addressing significant performance concerns such as procedural adherence, work control and tagging problems, ineffective communications and team work between organizations, continued weaknesses in identifying and correcting performance problems, and poor self-assessment and quality verification. In an effort to correct these management problems, in the past two years the licensee has replaced the three unit directors and the vice president of Millstone Station. Additionally, a corporate re-engineering in early 1996 eliminated the position of vice president of Millstone Station, and assigned individual corporate vice presidents to be responsible for overall utility operations, work services, technical services, and safety and oversight at all five NU plants. The vice

president of nuclear operations retired and was replaced early in 1996 by an individual from the Seabrook organization. The vice president of nuclear operations reports to NU's president.

Starting in November 1994, Unit 2 underwent a 9 month outage, during which numerous performance improvement initiatives were implemented. Key to improving its performance was overcoming the past failure of the operations department to drive or set plant priorities. There have been recent positive examples of operations assuming ownership of activities that affect plant operations. The plant came out of the extended outage and operated well until the recent shutdown and current outage.

The Unit 1 plant staff has not yet shown the same level of recognition and depth of response to their deficiencies as Unit 2, although the recently appointed unit director is aware of the challenges before him. Unit 1 is currently in a refueling outage that started in November 1995. It is noteworthy that there have been few performance problems to date during the outage. However, problem areas, particularly in maintenance, include quality of procedures, procedure adherence, accountability, and work control practices. The effect of system engineers onsite is the identification of numerous design deficiencies. More than 20 LERs related to design problems have been issued over the past 2 years, but a lack of a questioning attitude has been displayed in resolving these issues. The material conditions of the radwaste storage and processing system in Unit 1 was very poor, especially tanks and piping. The NRC issued escalated enforcement for this issue and the licensee has mounted a major effort to correct the condition of the system.

In January 1996, the regional office reorganized NRC staffing at the Millstone Station, changing from one senior resident inspector (SRI) and four resident inspectors (RIs) for all three units, to one SRI and one RI for each unit, all of whom report to an onsite NRC SES-level manager. Additionally, the agency established a senior executive as the Director, Millstone Oversight, and an NRR Project Directorate devoted exclusively to the Northeast Utilities sites.

Since the last SMM, at which all three units were placed on the Watch List, the agency has initiated several major efforts to understand in general the extent of the problems at the Millstone Station.

NRR issued letters to Millstone Units 1, 2, and 3 and Haddam Neck under 10 CFR 50.54 (f) requiring each unit to affirm their compliance with all of the rules and regulations and their licensing bases. The letters were sent as a result of the findings regarding the spent fuel pool at Millstone Unit 1, and because the licensee was not operating the facility in accordance with the licensing bases (this is discussed in an Inspector General's investigation finding). For Millstone Units 1 and 2, the affirmation must be made seven days before the units are restarted. For Millstone 3 and Haddam Neck, the letters required the affirmation for the plans and schedules for completing a comprehensive licensing bases review. NRR is conducting a major team inspection covering all four units to determine the scope of the licensee's noncompliance with the licensing bases, and the veracity of any affirmation

made by it. Subsequently, based on the results of a licensee internal audit, the NRC became aware of similar potential deficiencies at Millstone Unit 3 and issued an amended version of the 10 CFR 50.54 (f) letter that now requires the licensee to provide its affirmation response 7 days before the restart of that unit.

The 10 CFR 50.54 (f) letters are currently the critical path items for restart of all three Millstone units. Further, the licensee is having great difficulty setting the boundaries of its review and determining the scope of its effort.

In parallel with the licensee's efforts in response to the 10 CFR 50.54 (f) letters, NRR with regional support is processing a 10 CFR 2.206 petition dealing with the licensee willfully and knowingly offloading the full reactor core in violation of the licensing bases, which limited off loading under normal conditions to 1/3 of the core delayed for 150 hours after shutdown. The petitioners assert that the licensee misrepresented facts to the NRC in order to gain approval for license amendments. The Inspector General's investigation disclosed that the licensee had routinely offloaded the full core in less than 150 hours. There is also an Office of Investigation's (OI) inquiry into the wrongdoing aspects of this petition.

Release

On January 11, 1996, the licensee reduced its work force as part of its re-engineering and down sizing. Several of the employees who were discharged had previously supplied allegations to the NRC or were participants in other employee allegations. NRC management commissioned a team, led by the Office of the General Counsel, to determine if the licensee had proper justification to layoff past allegers.

The region and NRR have determined that Millstone Units 1 and 2 should be restarted under NRC Manual Chapter 0350. There are no orders or confirmatory action letters restraining the licensee from restarting any of the three units; only the 10 CFR 50.54 (f) letters are outstanding. Once the utility believes it complies with the conditions of the letters, they have a clear path to restart the units.

II.a. FUNCTIONAL AREA ASSESSMENTS: MILLSTONE UNIT 1

Operations

A. Assessment

Refueling outage (RFO) 15 began on November 4, 1995. The licensee planned a full core off load so that components located in the reactor vessel could be nondestructively tested. Off loading the full core had been the practice almost exclusively, although the final safety analysis report (FSAR) stated that full core off loads would be performed only in abnormal circumstances. The licensee had performed a 50.59 evaluation for the 1994 (RFO 14) full core

off load. Licensee engineers raised questions whether sufficient backup cooling was available to the spent fuel pool when the core was fully off loaded.

Based on these concerns, the licensee modified its shutdown cooling so that both trains of shutdown cooling were available as a backup to the spent fuel cooling system. The licensee submitted a license amendment to allow full core offloads as a routine option. This license amendment was approved by NRR on November 9, 1995. The license amendment was contested and is under review by the Atomic Safety and Licensing Board. The licensee began moving fuel on November 12, 1995. Subsequently, the group that was contesting the amendment did not tender issues within the designated submittal time and the staff will request the hearing be terminated. A 10 CFR 2.206 petition was filed regarding the operation of the Unit 1 spent fuel pool, which questioned whether some of the above practices actually constituted wrongdoing or intentional violations of NRC regulations. These [illegible] have been evaluated by OI.

An inspection at Unit 1 in 1995 noted that NU did not demonstrate the willingness and ability to promptly and critically address operability determinations, questions of component reliability, or considerations of system design basis. Consequently, the responsibility for operability determinations and safety evaluations was at times fragmented, and operability determinations were subjected to lengthy analysis or legalistic justifications.

The plant operators showed mixed performance in actual plant operations. They are highly experienced and operate the reactor professionally; however, they appear to be employing outdated techniques and methods to cope with the current NRC and industry standards and practices.

Unit 1 is currently shutdown and defueled. The critical path item for restart is its response to the 10 CFR 50.54 (f) letter, which requires an extensive design bases review, correlation with the updated FSAR, and assurance the design bases were translated into the technical specifications and operating procedures. This is necessary because of the age of the plant and its being licensed before the General Design Criteria and 10 CFR 50, Appendix B, applied. The licensee is currently projecting restart in late 1996.

B. Basis

- Corrective actions concerning a Notice of Violation for control room habitability issues were not comprehensive. Incomplete actions were identified in several areas, including root cause assessment, procedures, and training in the use of self-contained breathing apparatus.

- The recurrent alarming of the fire annunciator during surveillance testing of the diesel fire pump represented a willingness of operators to accept degraded conditions.

- A gas turbine fuel forwarding pump was determined to be inoperable on several occasions since 1994. The continued inoperability of the pump further demonstrates operators' accepting degraded plant equipment and not challenging unacceptable and informal operability determinations.

- The development and implementation of the shutdown risk plan for the outage was generally effective. The independent assessments of shutdown risk were thorough and provided good insights, which generally resulted in enhancements. Management oversight was generally good and ensured adequate defense in depth for key safety functions and overall plant risk.

- The operations staff failed to review the applicable operating procedures after learning of a core cooling bypass event at another plant. Procedures were subsequently determined to be inadequate. This deficiency was considered to be continuing evidence of overall poor quality of procedures, although once the problem was discovered, it was corrected promptly.

- The licensee performed an incorrect operability determination of the standby gas treatment system. In addition, each train of the system was not being independently tested in its design basis accident condition in that both trains of reactor building isolation dampers were shut during the secondary containment draw down test. This is an example of inadequate implementation of regulatory requirements.

- Several issues with operator performance following the shutdown were identified, although operator performance and supervision were good during the shutdown. The operations staff failed to prevent work (repacking of the recirculation loop valves) that had the potential for draining the reactor vessel while fuel was being moved. Also, the licensee did not have an adequate process to ensure all applicable technical specifications were implemented during the refueling outage.

- Operator actions to raise reactor pressure above the initial pressure assumed in the accident analysis reflected a lack of understanding and respect for the plant's design basis. In addition, the licensee failed to implement prompt procedure changes when the lack of procedure control over operating pressure was identified on two prior occasions.

- The shutdown for RFO 15 was well controlled. Shutdown procedures were properly implemented and the response to annunciators was good. Supervisory oversight was excellent.

- Infrequently performed tests and evolutions (IPTE) were performed properly and in a step by step manner. However, the IPTE procedures were improperly classified as general use procedures. This misclassification could have caused improper performance of the procedures.

- Fuel movement and overall refuelling activities were well controlled. Fuel movement was performed in a step by step process. Communications were excellent.

C. Plans

See Section III, Future Activities

Maintenance

A. Assessment

NU has acknowledged a lack of success to date in overcoming problems with the quality of procedures, procedural adherence, and poor work control and planning. These problems were further confirmed by the NRC during inspections of Unit 1's maintenance. These problems and the licensee's approach to correcting these deficiencies were discussed at a management meeting between NU and the NRC in late August 1995. Successful operation continues to be attributed to the quality and diligence of individual performance, rather than to programmatic effectiveness. NU's second major revision to the work control program within 2 years was unsuccessful at Units 1 and 2, and there is growing concern with the effectiveness of the procedure review and upgrade processes to correct deficient procedures. NU initiated new task assignments, managed at the site vice president level, to revisit these areas with the goal of obtaining a consistent, site-wide approach.

B. Basis

- Plant design calculations contain an assumption that a maximum of 20 percent of the total flow of the standby gas treatment system (SGTS) comes from the spent fuel pool floor. However, flow measurements of the reactor building HVAC ducting taken November 10, 1995, indicate that the actual flow was approximately 30 percent of the total SGTS flow. The SGTS' charcoal filters have the potential to be affected by moisture when spent fuel pool temperatures rise above 125 degrees F.

- The flow rates of reactor building component cooling water to the shutdown cooling system heat exchangers have exceeded the design flow rate, which is 1800 GPM per heat exchanger. During testing on November 8, 1995, it was discovered that heat exchanger "A" had a flow rate of 2500 GPM and heat exchanger "B" had a flow rate of 2700 GPM.

- The operations staff failed to test the SGTS as required by the Technical specifications following painting in a ventilation zone.

- A review of completed Generation Test Services (GTS) work orders found that approximately 85 percent were deficient. Problems existed with the adequacy of retests, quality of the work instructions, poor documentation of work performed, and one case in which work was

performed outside the job scope. In addition, the licensee's work observation program did not appear as effective a tool as intended. Further, there were several examples in which Adverse Condition Reports (ACRs) were not initiated to document unacceptable conditions, such as not meeting the as-found acceptance criteria during breaker maintenance. Based on the numerous examples of poor or inadequate retest criteria specified in the work orders, it appears that an unacceptably low standard for post maintenance testing has been established.

C. Plans

See Section III, Future Activities

Engineering

A. Assessment

NU's consolidation and unitization of engineering resources at the site has been completed. Recent observations have revealed a more responsive staff approach to operational issues. However, inconsistencies in the quality of engineering efforts are evident at each unit, indicating that further improvement in attention to detail and the use of conservative engineering principles are warranted. Unit 1 has issued approximately 20 LERs for design-related problems identified over the last 2 years; the NRC is escalating these issues as a major concern. Since the advent of the 2.206 issue and the 10 CFR 50.54 (f) reviews, numerous additional design related event reports have been made to the NRC.

B. Basis

- During November 1994 all six reactor coolant safety relief valves (SRVs) were removed and repaired due to excessive seat leakage. Testing of the six SRVs during a forced outage in July 1995 revealed that all six had again drifted high by an average of 7 percent. NU installed a pressure switch modification on each of the SRVs during the current refueling outage. However, the use of an electrical switch backup for the mechanical SRVs may not be in accordance with code. The need for licensing action to address the modification is an ongoing startup concern and possibly may be an unreviewed safety question. The NRC continues to monitor and evaluate this technical concern.

- There have been several 50.72 event reports concerning design discrepancies. All discrepancies are under evaluation by the licensee. Some examples include the following:

    * SGTS may not be capable of performing its intended function during certain conditions. Some reactor building ventilation fans may not trip as required under all conditions, causing flow rates in excess of 1100 SCFM. With flow rates in excess of 1100 SCFM, gas

holdup time in the charcoal beds is insufficient to permit acceptable radionuclide decay (January 4, 1996).

* The seismic anchorage for the emergency diesel generator day tank is inadequate (January 11, 1996).

* Parameters referenced in the FSAR list certain values for initial conditions for transients. The plant may have operated outside these design basis limits during the last operating cycle. Examples of limits that could have been exceeded were plant pressure, core thermal power, reactor core flow, and reactor steam flow (January 25, 1996).

- Test engineers directing special procedures in the plant exceeded the allowed overtime limits without adhering to the guidance contained in the administrative procedure governing the overtime policy. When acting in this capacity, engineers are directly responsible for the performance of the test and provide guidance to assigned plant personnel. As such, they are considered key maintenance personnel and are required to adhere to the overtime policy.

- Continuing problems with the performance of the SGTS was caused by inadequate understanding of the system's design details and inadequate system testing. These inadequacies contributed to the licensee incorrectly determining that the system remained operable with outside temperature greater than 20 degrees F, which resulted in the system being inoperable on several occasions between July and November 1995.

- Good engineering principles were demonstrated in the design and implementation of the loss of normal power modifications installed during the current refueling outage. The design effort was commensurate with the complexity and importance of the task, engineering demonstrated good performance in developing the criteria to maintain control over the installation and testing of the design changes, and the design change implementation was performed in a professional, systematic, and controlled manner. Further, the revised design received the necessary reviews and was supported by good safety evaluations. Appropriate controls were maintained during installation to assure the availability of emergency electrical power for the current plant conditions.

C. Plans

See Section III, Future Activities

Plant Support

A. Assessment

(Note: The Plant Support functions are sitewide at Millstone; therefore, this discussion applies to all three units.)

The NRC has noted the very poor condition of the radwaste processing system at Millstone, particularly at Unit 1. While the condition of the system did not constitute a radiological release concern, it is an ALARA concern for the radwaste system operators. The condition of the system is contrary to NU's attempts to reduce the volume of radwaste generated, and is representative of poor maintenance and a disregard for operating the plant in accordance with the FSAR description. The radwaste system at Millstone Unit 1 was the subject of escalated enforcement.

B. Basis

- Generally effective programs for radiation protection were observed. The training of plant health physics technicians and radwaste handling personnel was especially noteworthy.

- A violation of NRC requirements was identified regarding the operation of the Unit 1 liquid radwaste system. The long-standing leakage from various equipment, the deposit of radioactive spent resins on the floor of a radwaste tank cubicle, and the poor material conditions of the system represented an operational condition inconsistent with that described in Unit 1's updated FSAR.

C. Plans

See Section III, Future Activities

II.b. FUNCTIONAL AREA ASSESSMENTS: MILLSTONE UNIT 2

Operations

A. Assessment

Unit 2 restarted after an extended 10 month outage on August 4, 1995. A pipe rupture on August 8, 1995, delayed its return to full power operation. After repairing the piping, the unit returned to full power on August 16, 1995, and remained at power without incident (except for some minor power reductions due to mussel fouling in the circulating water system) until December 14, 1995. On December 14, the reactor was conservatively shutdown when the structural integrity of a leaking 2-inch, spring-loaded check valve, 2-CH-435, could not be determined. The check valve was designed to provide thermal relief on the charging side of the regenerative heat exchanger. The cause of the small amount of leakage was determined to be the result of an original construction deficiency on the valve cover torquing. The valve was repaired and the unit returned to power on December 19, 1995.

On February 20, 1996, the high pressure safety injection system (HPSI) was declared inoperable due to the potential for clogging the HPSI throttle valves from debris during containment sump recirculation. The plant staff noted a

nuclear network issue at another reac r that dealt with a similar issue. The plant was placed in cold shutdown after the licensee determined that the unit's sump strainers were not properly sized relative to the opening of the throttle valves; as required by Technical Specification 3.03. Following the shutdown, the licensee decided to remain in cold shutdown and start a previously scheduled midcycle surveillance outage 2 months early.

Senior Region I managers were concerned that the cause of a number of the performance problems at Unit 2 was the failure of the Operations department to drive or set plant priorities. NU has noted this problem for some time and has been focusing management attention in this area. Much effort has been expended to re-vitalize the shift supervisor position at Unit 2. Extended re-qualification cycles have focused on recognition and ownership of existing performance weaknesses, remediation strategies, and team building.

The Unit 2 Operations Manager was replaced twice; once in July 1994 and most recently in March 1996. The need for improved procedural adherence, attention to detail, self-verification and configuration controls is being emphasized site-wide to address the continuing occurrence of errors and minor events, which detract from otherwise good operational performance. Unit 2 operators are beginning to demonstrate more ownership of the plant. The NRC has noted recent, positive demonstrations of operator ownership since the plant's restart from the extended outage. The licensee made substantial improvements in the control room configuration during the extended outage to enhance operator human factors.

In March 1996, Unit 2 along with Unit 3 and Haddam Neck of the Northeast Utilities nuclear units, received a 10 CFR 50.54 (f) letter requesting that the licensee reaffirm its compliance with the licensing bases. This letter was generated as the result of discrepancies in the licensing bases and other design inadequacies discovered in the last three months at the other units.

B. Basis

- On February 20, 1996, the licensee began shutting down Unit 2 when both trains of the high pressure safety injection system (HPSI) were declared inoperable due to the potential clogging of the HPSI discharge throttle valves during the recirculation phase following a loss of coolant accident (LOCA). Taking precautions to avoid exceeding the reactor coolant system cool down rate caused the operators to exceed the technical specification (TS) required time to reach cold shutdown.

- A power reduction to 55 percent to support a main feedwater valve repair was well controlled, and there was good procedure usage by operators. There were two notable examples of operators demonstrating a good questioning attitude and playing a lead role in decisions affecting plant operations and safety. However, there were several concerns that overall performance by operators is varied.

- The overall operator training and requalification program was excellent, as evidenced by challenging written and operating examinations, higher training expectations, and a good training feedback and monitoring program. Management oversight and involvement in the licensed operator training and requalification programs were effective in establishing high standards of operation for the observed crews.

- The plant was shut down on December 14, 1995, when the structural integrity of a leaking charging system check valve could not be determined. The licensee's decision not to attempt a valve repair with the unit on line demonstrated a good safety perspective.

- On December 17 1995, the technical specification limit for the pressurizer heat up rate of 100 degrees F in any one hour was exceeded. The excessive heat up rate occurred because during a plant heat up non-condensable gasses at the top of the pressurizer inhibited a uniform heating of the pressurizer metal. The top of the pressurizer heated up rapidly when the gasses dispersed upon initiation of pressurizer spray. On January 24, 1996, plant data revealed that the RCS heat up rate was also exceeded during the December 17 heat up. The RCS heat up rate had also been exceeded on July 28, 1995. A licensee event review team is currently reviewing the three events to determine commonality, root cause and corrective actions.

- On January 8, 1996, both trains of service water were rendered inoperable by an ice plug in a common backwash line for the service water strainers. When the condition was discovered, the Shift Manager failed to declare the service water system inoperable and licensee management's response to the event was inadequate. The plant was in this condition for 5 hours. This incident is being processed for escalated enforcement action.

C. Plans

See Section III, Future Activities

Maintenance

A. Assessment

NU's second major revision to the work control program within 2 years was unsuccessful at Units 1 and 2. There is recent evidence that the effectiveness of the procedure review and upgrade processes to correct deficient procedures is meeting with some success. However, instances of operators performing steps out of sequence and deficient surveillance procedures continue. NU initiated new task assignments, managed at the site vice president level, to revisit these areas with the goal of obtaining a consistent, site-wide approach.

B. Basis

- The licensee failed to update the anticipated transient without scram (ATWS) system surveillance to reflect modifications that changed the position of four contacts from normally closed to normally open. In addition, the licensee had performed the surveillance satisfactorily even though 24 separate verifications could not be completed in accordance with the procedure.

- An NRC review of upgraded maintenance procedures found their technical quality to be good.

- An atmospheric steam dump inadvertently opened when steps in the associated surveillance procedure were performed out of order.

- Due to inadequate focus on the status of an inoperable effluent radiation monitor associated with the waste neutralization sump tanks, "best efforts" were not made to return the radiation monitor to service as required by technical specifications.

C. Plans

See Section III, Future Activities

<u>Engineering</u>

A. Assessment

NU completed its consolidation and unitization of engineering resources at the site. Recent observations have revealed a more responsive staff approach to operational issues. However, inconsistencies in the quality of engineering efforts are evident at each unit, indicating that further improvement in attention to detail, better definition of the design bases, and the use of conservative engineering principles is warranted.

B. Basis

- There was a basic lack of understanding by the licensee of the functioning of the Unit 2 hydrogen monitors, post accident sample system (PASS), and containment radiation monitors. As a result, both trains of the hydrogen monitors were inoperable because a vacuum regulating valve would prevent air flow through the monitor cell when containment pressure was low.

- The licensee also failed to satisfy its licensing basis and design basis regarding the time required to place the hydrogen monitors in service and draw a containment air sample using PASS.

- Preventive maintenance and surveillance of the hydrogen monitors did not adequately ensure the equipment was properly adjusted, calibrated, and maintained. The licensee's corrective actions to resolve a single failure vulnerability associated with the containment isolation valves to the hydrogen monitors and PASS were not effective.

- A conservative and comprehensive operability determination for the pressurizer spray piping and support system found the assumed snubber failure caused no deleterious effect on the system piping. Millstone has provided appropriate engineering effort in comprehensive and timely response to resolution of the unresolved items.

- The licensee identified that the design basis ambient temperature for the intake structure would be exceeded following the loss of three non-vital ventilation fans, but then failed to promptly correct this deficiency.

C. Plans

See Section III, Future Activities

II.c. FUNCTIONAL AREA ASSESSMENTS: MILLSTONE UNIT 3

Operations

A. Assessment

Operator performance during this period, although better than Units 1 and 2, was mixed. In general, control room evolutions were well performed. Operators handled a feedwater water hammer event well and performed a well controlled shutdown of the plant. However, performance by operations for more routine circumstances showed inattention to detail. Events such as an unplanned dilution of the RCS, problems operating a reactor coolant isolation valve, and failure to reconnect a battery charger cable are examples of this.

The licensee has failed to take effective corrective actions to resolve past problems, which led to the unplanned dilution, the feedwater water hammer, and a seal leak on the volume control tank. These isolated instances indicate lapses in a questioning attitude among operators for unusual occurrences.

The need for improved procedural adherence, attention to detail, self-verification and configuration controls is being emphasized site-wide to address the continuing occurrence of errors and minor events, which detract from otherwise good operational performance. Unit 3 operators appear to be the most successful in this area, with very good operational performance recently.

The unit had a good run after starting up on September 21, 1994, from a plant trip, and then shut down for a refueling outage on April 14, 1995. After a successful refueling outage, which lasted 51 days, the unit was restarted on June 6, 1995, and returned to full power operations. However, the unit experienced recurring leaks from check valves in the auxiliary feedwater line to the "D" steam generator. The leaks required the operators to periodically cycle an auxiliary feedwater pump to cool the piping and the containment penetration. On August 6, 1995, one check valve in this line was cut out and replaced. Subsequently, controls for reseating check valves were coordinated with auxiliary feedwater pump surveillance runs. The problem has not recurred since December 1995, but further analyses of the piping, replacing other check valves, and design modifications are deemed necessary to implement a permanent solution.

On November 30, 1995, Unit 3 was shutdown to investigate and repair a leaking check valve in the reactor coolant system (RCS). During this shutdown, two cracked socket welds were discovered in the piping for RCS flow instrumentation. The licensee cooled down the reactor to repair the defective welds and the leaking check valve. The unit resumed full power operations on December 15, 1995. The unit generally remained at full power but on several occasions reduced power to backwash fouling of the main condenser bays. On March 30,1996, the licensee executed a technical specification shutdown because of the inability of the Target Rock containment isolation valves in the auxiliary feedwater system to seal against pressure in the reverse direction (i.e., pressure coming from the containment side). Subsequent to the shutdown, NRR issued an amended 10 CFR 50.54 (f) letter requiring the licensee to respond 7 days before they restart the reactor.

B. Basis

- On November 10, 1995, the RCS was inadvertently diluted through the chemical and volume control system, causing reactor power to increase above 100 percent for 35 minutes.

- Unit 3 was safely operated with evidence of good command and control of plant activities by the licensed operator staff, as well as an awareness of good self-checking practices in the field.

- The procedural controls for restoring a feedwater heater were found to be lacking the proper guidance and criteria for successful restoration. Although the equipment is not safety related, the licensee's inability to correct a past problem in this area led to recurring water hammers.

- While the implementation of the adverse condition reporting program at Unit 3 has been rigorous, the number and scope of the identified problems continues to merit upper management attention. The Unit 3 Director has demonstrated strong involvement and a safety conscious attitude in the resolution of the identified concerns.

- Unit 3 was safely operated in accordance with approved procedures and regulatory requirements. Improvements to the control room complex were initiated during this period. The shift managers demonstrated good command and control of the ongoing construction activities to prevent adverse impact upon safe plant operation.

C. Plans

See Section (II, Future Activities

Maintenance

A. Assessment

Unit 3 maintenance management provided good oversight of work activities, and work was generally performed well. The technical skills of the craft were very good, although lapses in the oversight of work controls caused occasional problems. At Unit 3 the lapses were attributed to procedures that were either deficient or not followed.

B. Basis

- Maintenance activities associated with repairs of various reactor coolant system leaks were performed adequately. Lack of maintenance on several components contributed to an unplanned dilution of the RCS and a spill of approximately 500 gallons of primary coolant: the letdown system high temperature divert valve and the remote operators for the "A" seal injection filter isolation valves.

- Preventive and corrective maintenance and surveillance activities were properly implemented with evidence of good interdepartmental coordination. The use of bypass/jumper controls for the areas reviewed was found to be appropriate.

- The licensee's conduct of engineering reviews to verify that other generic industry problems were not adversely impacting Unit 3 operations was noted to be proactive and appropriately focused

- Selected plant design changes and the related design control practices were noted to have been properly authorized and verified to assure adequate qualification, safety evaluation, and FSAR revision processing. Configuration management, system design details, and component conditions were found to be adequately controlled.

- Preventive and corrective maintenance activities were adequately controlled. A surveillance test for the system restoration of the "B" emergency diesel generator was properly performed and well coordinated between the field and the control room. NRC review of the vital electrical bus power configuration, in light of ongoing troubleshooting

activities, was found to be acceptable and consistent with the licensee's determination that the affected power supplies and components were operable.

C. Plans

See Section III, Future Activities

Engineering

A. Assessment

NU completed its consolidation and unitization of engineering resources at the site. Recent observations have revealed a more responsive approach by plant staff to operational issues. However, inconsistencies in the quality of engineering efforts are evident, indicating that further improvement in attention to detail and the use of conservative engineering principles is warranted. For example, engineering recently identified that the design temperature for the piping in Unit 3's recirculation spray system could be exceeded during certain scenarios. This condition was originally identified by the architect engineer before the plant was licensed in 1985; however, corrective actions were never initiated until recently.

B. Basis

- The recirculation spray system (RSS) at Millstone Unit 3 is designed to provide ECCS flow during the recirculation phase of a LOCA; i.e., when the source of cooling water is the containment sump instead of the refueling water storage tank (RWST). The spray headers are designed for a temperature of 150 degrees F. On November 13, 1985 (NOTE: Unit 3 received its low-power license on November 25, 1985), the architect engineer for the unit, Stone and Webster, documented a concern regarding the loss of service water to an RSS heat exchanger, which would allow uncooled (greater than 150 degrees F) containment sump water to be injected through the RSS piping. Stone and Webster recommended either of three options; (1) pipe support modifications, (2) automatic circuitry to isolate the RSS train with loss of service water, or (3) operational procedures to direct the operators to trip the affected RSS train upon loss of service water. Currently, none of the three options have been implemented.

- Licensee management displayed a lack of conservative safety perspective in not validating the conditions that existed with a leaking 10-inch unisolable check valve in the RCS. The associated operability determination was based on engineering judgment and assumptions, and was not entirely accurate. Concerns developed regarding whether the leak was coming from the RCS pressure boundary, and whether the actual conditions of the valve supported the operability determination assumed condition of the valve cover bolting. On November 30, 1995, the

licensee placed the plant in hot shutdown to investigate the leak, and subsequently shut down to cold shutdown when additional, unrelated leaks in the RCS pressure boundary were discovered.

- The licensee's investigation of failures in RCS small bore socket welds, identified during the Unit 3 outage, was broad, addressed the generic implications of socket weld failures at different locations, and provided comprehensive corrective actions.

- Selected plant design changes and the related design control practices were noted to have been properly authorized and verified to assure adequate qualification, safety evaluation, and processing of FSAR revisions. Configuration management, system design details, and component conditions were found to be adequately controlled during plant inspections and tours.

- During the outage, the licensee also identified an active coolant leak from the seal housing on the "D" reactor coolant pump (RCP). The leaking seal gasket and all 18 seal housing bolts were replaced. The carbon steel alloy bolts, which are highly susceptible to attack from wetted, boric acid, have the potential for corroding. Subsequent inspections resulted in the conservative decision to replace all seal housing bolts on the three remaining RCPs, and the seal gasket on the "B" RCP. Licensee actions to determine the cause of RCP seal housing leaks were comprehensive and identified a design problem, which was promptly corrected to preclude future wetting of seal housing bolts. Improved management performance in following through with appropriate corrective actions was noted.

C. Plans

See Section III, Future Activities

III. FUTURE ACTIVITIES

The current inspection schedule was developed by the Millstone Oversight Team, with inspections focusing on the following areas:

- Procedure adherence at all three units by continuing resident and specialist inspector inspection, and Unit 1 maintenance and allegation followup activity.

- NU's root cause, self-assessment and corrective action programs.

- Operator use of control room procedures, procedural adherence, and processing of work requests. (This area was assessed by a startup team in June 1994, the Readiness Assessment Team Inspection (RATI) in May 1995, and continues to be reviewed by ongoing resident and specialist inspector activity.)

- NU's Nuclear Safety Concerns Program (NSCP), and the status of NU corrective actions in this area.

- The adequacy of operational ownership of plant activities at Unit 2.

DATA SUMMARY

I. PRA

I.a. UNIT 1

A. PRA Insights

Millstone 1 is an early BWR 3, Mark I with no HPCI or RCIC to mitigate high pressure transients or LOCAs. Instead, Millstone 1 is equipped with an Isolation Condenser (IC) and a Feedwater Coolant Injection System (FWCI), which receives emergency power from a gas turbine, to mitigate high pressure transient events. The FWCI system can be also be used to mitigate small LOCAs.

Although the original Millstone 1 PRA and its early updates indicated that Large LOCA was the dominant contributor to core damage, the IPE reports a reduced CDF estimate which is dominated by Station Blackout (SBO). In the event of Loss of Normal AC Power, Unit 2 can cross feed emergency power to Unit 1. In order to comply with the SBO rule, the licensee installed weather protection around the emergency cross feed areas and upgraded associated relays. In the event of a prolonged SBO, the IC is expected to provide a reliable means of core cooling. The site has experienced several extended losses of offsite power (LOOPs) that were caused by hurricanes. The licensee has adopted a policy of shutting down the reactors at the site on the approach of hurricanes, which should reduce the SBO contribution to risk.

Published PRAs provide a strong indication of the risk significance of Service Water Systems (SWS), the ultimate heat sink for the safety related systems. The Millstone 1 SWS is prone to cracking/erosion problems, and the licensee continues to replace service water piping sections as necessary every refueling outage.

B. PRA Profile

A full-scope PRA that predates the IPE requirements was performed by the utility for Millstone 1 and has been reviewed by the staff. The full-scope PRA estimated the CDF to be approximately 1E-4/yr, with the largest single contribution coming from fires, followed by large LOCAs. The dominant risk areas for fires were the cable vault, the switchgear area, the feedwater area, and the reactor building.

The licensee responded to GL 88-20 by submitting its IPE dated March 31, 1992. The IPE reported the CDF to be 1.1E-5/yr from internal events (including internal flooding but excluding fires). The IPE includes changes to plant hardware, additional EOPs and refined engineering analyses that affect success probability estimates.

The contributions to CDF reported in the IPE by initiating event group are listed below:

Initiating Event Category	CDF (/yr)	% CDF
Loss of Normal Power (SBO)	8.1E-06	73.2%
Small LOCA	4.0E-07	3.6%
Inadvertent SRV Opening	4.0E-07	3.6%
Internal Flooding	2.5E-07	2.5%
Loss of Service Water	2.1E-07	1.9%
ISLOCA	1.3E-08	1.1%
Loss of Feedwater	7.0E-08	0.6%
Large LOCA	5.7E-08	0.5%
Medium LOCA	6.2E-09	0.06%
Other Reactor Transients	1.4E-06	12.8%

RES has completed its review of the Millstone 1 IPE submittal and issued an SER on October 5, 1993, and concluded that the licensee has met the intent of GL 88-20. It is important to note that fires, which were not included in the IPE submittal, contributed about 29% of a total CDF of 1E-4/yr in the Millstone 1 full-scope PRA. Therefore, the IPEEE covering the risk from fires, seismic, and other externally initiated events, which is scheduled to be submitted 12 months after the end of the Cycle 15 outage, may dominate the risk profile.

The licensee did not identify any major vulnerabilities with respect to core damage in performing the IPE. However, the licensee identified that drywell steel liner melt-through by molten debris following core melt and RPV failure is a major vulnerability with respect to containment performance. The licensee has made a modification to the LPCI heat exchanger so that fire water can be aligned via hoses to supply water for drywell spraying if a supply from LPCI pumps are not available. In the IPE, the licensee credited containment venting through its existing (not hardened) wetwell vent path and has estimated that providing a hardened vent would decrease CDF by <1%. Nevertheless, the licensee has later installed the hardened vent.

Major contributors to dominant core damage sequences in the IPE include emergency gas turbine generator unreliability, EDG failure to start, maintenance unavailability of the EDG and the gas turbine generators, failure of the diesel room coolers, and failure of the operator to establish isolation condenser cooling when needed.

The IPE estimated the following mean conditional source term release probabilities:

Early release:	0.34
Intermediate release:	0.28
Late release:	0.02

C. Core Damage Precursor Events

On the basis of the precursors identified by ORNL for 1994 (NUREG/CR-4674, Vols. 21 and 22) and the preliminary precursors for 1995 and 1996, the staff did not identify any precursor events for the site that have a conditional core damage probability (CCDP) of 1E-5 or greater. However, the following event was identified as an "interesting event" by ORNL:

> During testing on day 85 of a plant outage, the shutdown cooling (SDC) system was inadvertently aligned to the drywell spray system. Approximately 12,000 gallons of reactor water coolant inventory was sprayed into the drywell before operators identified and isolated the leakage pathway. Automatic isolation of the drain down should occur when the vessel reaches 132 in. above the top of the active fuel. This would require the closure of one of the three motor-operated valves upon receipt of a group III isolation signal. If the isolation signal did not occur, makeup via the core spray pumps would be possible. The LPCI pumps were unavailable since their breakers were racked out. If neither of these actions is successful, the vessel will be drained to the top of the active fuel in about 10 min. The drain down will continue until the SDC pumps lose adequate suction head. This event involved an intersystem LOCA at shutdown.

In addition, no recent events have been classified as "Significant Events" in the Performance Indicator Program.

I.b. UNIT 2

A. PRA Insights

Millstone 2 is an early CE NSSS plant with a large, dry containment.

The auxiliary feedwater system (AFW) has two motor driven AFW pumps and one turbine driven AFW pump. If the human error probability for operator failure to locally start the turbine driven AFW pump is increased by a factor of 10, the total IPE CDF increases by a factor of approximately 3%.

Should all feedwater be lost the feed and bleed capability at this unit is minimal. The Safety Injection (SI) pumps (used for high pressure injection) have a shutoff head of 1205 psig. The three positive displacement charging pumps, which would have to be aligned manually, have low capacities (44 gpm each) and provide little feed and bleed assistance. Since the two PORVs are small, operator action must be rapid to assure success in the event that bleed and feed is needed.

Station blackout is considered to be an important contributor to estimated CDF for many plants. In Millstone 2, however, the licensee indicated that the total SBO contribution (1.25%) to CDF is relatively low due to the diverse emergency AC power supplies including an alternate AC power source which is a cross-tie to either one of the Unit 1 emergency AC power sources. The cross-tie reduces the SBO contribution by about an order of magnitude. The alternate AC source can be made available within 1 hour from the onset of an SBO. During refueling outage 11, which ended in January 1993, weather protection was installed to the alternate AC components in order for Unit 2 to fully comply with the SBO rule. The station batteries have only a one-hour capacity, as reported in the FSAR. The Millstone site has been subject to several extended losses of offsite power due to hurricanes. The licensee has taken the position that it will shut down the reactor if a hurricane is in proximity, further reducing the SBO contribution to risk.

B. PRA Profile

A Level 1 PRA was completed by Millstone 2 in the fall of 1991. In response to GL 88-20, the licensee submitted their IPE, which consisted of an updated Level 1 PRA coupled with Level 2 PRA, dated December 30, 1993.

The IPE review has been completed by RES as documented in the SER dated March 29, 1996. While the staff concluded that the IPE submittal met the intent of Generic Letter 88-20, weaknesses in the Level 1 and HRA portions were identified. The licensee plans to update the IPE.

The IPE reported that the core damage frequency (CDF) due to internally initiated events is 3.4E-5/yr including internal flooding. The contributions to CDF by initiating event group are listed below:

Initiating Event Group	CDF(/yr)	% of Total CDF
Loss of Offsite Power	8.4E-06	24.6
Transients	6.1E-06	17.8
Loss of DC Bus A	3.9E-06	11.4
Loss of DC Bus B	3.9E-06	11.4
Steamline Break	2.9E-06	8.5
Large LOCA	1.7E-06	4.8
Small LOCA	1.6E-06	4.8
Small-small LOCA	1.5E-06	4.3
Medium LOCA	1.3E-06	3.7
Loss of Service Water	9.7E-07	2.8
Loss of Vital AC Panels 10&30	8.4E-07	2.5
SGTR	5.2E-07	1.5
Main Feedline Break	2.4E-07	0.7
Internal Flooding	2.0E-07	0.6
ISLOCA	6.6E-08	0.2

The results from the original Level 1 PRA led to several plant changes designed to decrease the overall CDF, and these changes have been reflected in the updated Level 1 PRA. Some of the changes resulted in a decrease of AFW unavailability by increasing the ability to isolate an intact SG from a faulted SG. Another change required additional testing of LPSI check valves for reverse flow, thereby increasing their availability and reducing the possibility of failing LPSI cold leg piping due to overpressurization.

Based on the screening criteria in the IPE, one vulnerability sequence had been identified. The sequence involves reactor coolant pump thermal barrier failure leading to possible overpressurization of reactor building closed cooling water and a small ISLOCA. This sequence is outside the design basis of the plant. The licensee plans to address this vulnerability in April, 1997 by installing relief valves in the reactor building closed cooling water system to limit pressure buildup. These valves will discharge into the containment, thus significantly reducing the possibility of an ISLOCA.

The Level 2 analysis found that the conditional probability of early containment failure is approximately 9.7%, and is dominated by direct containment heating. The conditional probability of late containment failure is approximately 33.1%, and dominated by the basemat melt-through failure. The conditional probability that the containment remains intact is 57.2%.

The licensee has indicated that the IPEEE will be submitted in the fall of 1996.

C. Core Damage Precursor Events

On the basis of the precursors identified by ORNL for 1994 (NUREG/CR-4674, vols. 21 and 22) and the preliminary precursors for 1995 and for 1996, the staff did not identify any precursor events for the site that have a conditional core damage probability (CCDP) of 1E-5 or greater.

The following events have been classified as "Significant Events" for the Performance Indicator Program:

> In addition to accident sequence precursors, events involving loss of containment functions are reported in NUREG/CR-4674, vol. 21. The following event was classified in NUREG/CR-4674, vol. 21 as a containment-related event. On December 6, 1994, the licensee identified a release path from the Enclosure Building that would allow a direct discharge to atmosphere bypassing the charcoal filters following a LOCA. Specifically, a hydrogen analyzer cabinet and sample hood exhaust fan were found to take a suction on the enclosure building and discharge out the Millstone 2 Main Exhaust stack. This flow path had HEPA filters but lacked

charcoal filters. Given this release path, the licensee calculated that the site boundary thyroid dose would be more than twice the 10 CFR 100.11 limit in the event of a design basis accident, core melt, and subsequent fission product release. The concerns from this event were resolved through hardware modifications.

On January 26, 1995, Millstone 2 determined that both containment sump recirculation gate valves may experience pressure locking during a design basis LOCA and fail in the closed position. A recent assessment by the licensee indicated that these valves would have been able to perform their intended function; however, the staff has not been able to verify this. The failure of these valves would make the water source for the ECCS and the containment spray unavailable during the recirculation phase of the LOCA. If these valves had been inoperable at times in the past, then the estimated increase in CDF would be in the 1E-3/yr range, assuming that the recirculation function is not recoverable and that it is essential for mitigation of medium and large LOCAs. If recovery is possible, the estimate would be reduced. Opportunities for recovery might include manually opening a containment sump valve, or obtaining additional borated water from Millstone 3. Based on the licensee's recent determination that the valves were operable, the NRR/AEOD/RES Events Assessment Panel re-examined this event and kept it as a "Significant Event" based on programmatic weaknesses in the licensee's management of the plant.

Ic. UNIT 3

A. PRA Insights

Millstone 3 is a Westinghouse four-loop NSSS with a sub-atmospheric containment.

The RWST volume at Millstone 3 is approximately 1.2 million gallons which is larger than the a typical RWST volume at other PWRs. The large RWST volume allows a longer time before switchover to the recirculation mode is required for LOCA scenarios. The IPE submittal indicated, however, that for large and medium break LOCAs, failure of high or low pressure containment sump recirculation is a major contributor to the IPE CDF.

Loss of main feedwater and loss of feed and bleed cooling is important at Millstone 3. A sensitivity analysis performed by the licensee in response to an IPE review question found: 1) if no credit was taken for feed and bleed, the total IPE CDF would increase by a factor of 4; 2) if no credit was taken for main feedwater, the total IPE CDF would increase by a factor of 2.6; and 3) if no credit was taken for feed and bleed

cooling together with loss of main feedwater, the increase would be a factor of approximately 13 in the total IPE CDF.

Station blackout (SBO) sequences are not a large contributor to the CDF; however, these sequences are important contributors to large scale fission product releases. The licensee installed a new air cooled diesel generator during the 1993 refueling outage to comply with the SBO rule. This third diesel generator decreases the SBO risk.

The 1983 Probabilistic Safety Study (PSS) indicates that ISLOCA sequences contribute over 98% of the risk from early fatalities; however, ISLOCA sequences are small CDF contributors in the PSS and the IPE. One important ISLOCA release path identified in the PSS was via the RHR pump suction to the ESF building.

Failure of the safety injection accumulators appear in several top accident sequences in the IPE. This was due to the long test interval between testing of the discharge check valves. Based on PSA insights, partial stroke testing of the valves is performed every refueling outage.

Published PRAs provide a strong indication of the risk significance of service water systems, the ultimate heat sink for the safety systems. The Unit 3 service water system piping has experienced significant erosion/corrosion which has resulted in wall thinning and pipe leaks. During the 1992 outage, 60 percent of the small bore SWS piping was replaced. In the refueling outage which ended in Nov. 1993, most of the residual small bore SWS piping was replaced. SWS piping replacement and upgrades will continue on an as necessary basis, every refueling outage.

B. PRA Profile

The licensee completed a Level-3 PSS in 1983 for Millstone 3 that included internal and external events. The PSS was motivated by a 1981 NRC letter over concern that the Millstone site was a high population area and therefore might make those living in the area subject to disproportionate risk.

The licensee submitted its IPE and its IPEEE in August 1990. The internal events CDF is 5.6E-5/yr and the external CDF is 1.4E-5/yr. RES completed its review of the internal events portion of the IPE submittal and concluded in the SER dated April 2, 1992 that, with the installation of the third diesel generator, the licensee met the intent of Generic Letter 88-20. The IPE risk profile is listed below by initiating event:

Initiating Event	% CDF
Transients	
LOOP	9
Loss of 1 DC bus	7
Loss of 1 SW train	5
Steam line break outside containment	15
LOCAs	
Large LOCA	15
Medium LOCA	19
Small LOCA	4
ATWS	6
SGTR	2

The IPE profile for conditional containment failure probability given core damage is as follows:

Intact containment	80%
Late containment failure without sprays	11%
Basemat failure with sprays	4%
Late containment failure with sprays	3%
Basemat failure without sprays	<1%

The Millstone 3 IPE submittal concludes that there are no major severe accident vulnerabilities requiring immediate corrective action.

The licensee's IPEEE submittal has undergone working level review in RES and is now in the management concurrence chain.

C. Core Damage Precursor Events

On the basis of the precursors identified by ORNL for 1994 (NUREG/CR-4674, vols. 21 and 22) and the preliminary precursors for 1995 and for 1996, the staff did not identify any precursor events for the site that have a conditional core damage probability of 1E-5 or greater. In addition, no recent events have been classified as "Significant Events" in the Performance Indicator Program.

II. ENFORCEMENT HISTORY

6/94 - LETTER TO LICENSEE (EA 94-012): The action was based on an investigation conducted by the Office of Investigations (OI) that concluded that a licensee official had deliberately failed to provide complete and accurate information in a letter to the NRC dated November 13, 1989, regarding a harassment and intimidation issue. After a thorough review of the OI report and a detailed consideration of the matter, the staff concluded that no regulatory action was warranted. The NRC issued a letter to the licensee to emphasize the importance of submitting information that is complete and accurate in all material respects.

7/94 - CIVIL PENALTIES AND DEMAND FOR INFORMATION (Supplement I, Reactor Operations; and Supplement VII, Miscellaneous Matters; EA 91-127): The action was based on an investigation by the Office of Investigations and an inspection conducted by Region I, concerning two violations consisting of: (1) the deliberate delay in taking corrective actions for a condition adverse to quality (specifically a deliberate delay in the determination of the operability of the feedwater coolant injection system) that continued from June 1989 to November 1989, and (2) discrimination by an unit Engineering Manager in April 1990 against an engineer who was involved in the operability determination. Civil Penalties were issued to emphasize the importance of prompt resolution of potential safety concerns when they exist, as well as ensuring that appropriate controls exist to preclude discrimination of employees who raise such concerns. In addition, a Demand For Information was issued to the licensee demanding an explanation why the NRC should not issue an Order to modify their license to preclude the Engineering Manager from any involvement in future licensed activities at any of the licensee's facility. The civil penalty for the first violation was escalated to $120,000 due to the continuing nature of the violation that spanned a number of months in 1989. The civil penalty for the second violation (categorized at Severity Level II) was escalated for NRC identification, but was limited to the statutory maximum of $100,000. ($220,000)

8/94 - CIVIL PENALTY AND EXERCISE OF ENFORCEMENT DISCRETION (Supplement I, Reactor Operations; and Supplement VIII, Emergency Preparedness; EA 94-045, and EA 94-091): This action was based on two violations. The first violation consisted of three examples of failure to classify and notify NRC of events at Unit 2 that constituted Unusual Events under the licensee's Emergency Plan and Procedures. The second violation involved the failure to ensure that adequate shutdown margin requirements existed following the identification of an immovable control rod assembly in April of 1994. The violations were categorized in the aggregate as a Severity Level III problem. To emphasize the importance of appropriate classification and notification to the NRC of an Unusual Event and maintaining adequate shutdown margin to protect the health and safety of the public, a civil penalty was issued after escalation of 50% for both

identification and licensee performance and mitigation of 25% for the licensee's corrective actions. Enforcement discretion was exercised for a third violation that involved the compromise, since 1987, of the automatic primary containment isolation function for a postulated reactor water cleanup system line break. Discretion was exercised because the problem originated more than 7 years ago, it was not a willful violation, adequate corrective actions had been taken by the licensee, and it was licensee-identified. ($87,500)

2/95 - EXERCISE OF ENFORCEMENT DISCRETION (EA 95-004): This action was based on the licensee's identification on December 6, 1994 of an oversight in the original design of the discharge flow path for the hydrogen analyzer ventilation system which could have permitted excessive offsite doses in the event of a loss of coolant accident. Enforcement discretion was exercised because: (1) the condition was identified by the licensee's staff as a result of a healthy questioning attitude and was promptly reported to the NRC; (2) the condition existed since initial startup, was difficult to discover, and such identification was not likely by routine inspection, surveillance, and quality assurance activities; (3) comprehensive corrective actions were taken within a reasonable time period that involved an adequate root cause determination and a review for failures caused by similar root causes; and (4) the condition was caused by an old performance failure that was not reasonably linked to present day performance.

5/95 - CIVIL PENALTY (Supplement I, Reactor Operations; EA 95-031): The action was based on violations associated with the licensee's failure to identify that the containment sump recirculation valves were susceptible to becoming pressure locked shut in certain accident conditions. In particular, a contractor study had identified the valve vulnerability in an October 1994 report, but the licensee took no action until January 1995. The violations were characterized as a Severity Level III problem. Although mitigation was warranted for the licensee's corrective action, it was offset by the escalation for poor past performance. ($50,000)

6/95 - LETTER TO LICENSEE (EA 95-093): The action was based on a Department of Labor (DOL) complaint filed by a former Millstone worker against the licensee, and the subsequent Decision and Order issued by the Secretary of Labor on the case. The former worker alleged that he had been discriminated against because the licensee offered him a monetary settlement in exchange for his agreement to restrict his participation in future regulatory proceedings. The Secretary of Labor ruled in favor of the former worker; however, the staff decided that enforcement action was not warranted because of the age of the issues (the events occurred in 1989), the fact that several other significant enforcement actions and civil penalties have been issued to the licensee for violations involving harassment and intimidation, and the Secretary of Labor had ordered a remedy for the former worker.

12/95 - SEVERITY LEVEL III VIOLATIONS (Supplement I, Reactor Operations; EA 95-177): The action was based on two Severity Level III violations involving (1) a modification in 1976 that resulted in an existing single failure vulnerability in the loss of normal power logic that would have prevented both emergency power sources from properly starting and sequencing the loads and (2) the inoperability of the standby gas treatment system under certain conditions from 1968 until June 21, 1995. A civil penalty was not proposed in either case to encourage prompt identification and correction of violations. Because these were not the first escalated actions within 2 years, the NRC considered whether credit was warranted for identification and corrective action. The civil penalties were fully mitigated in both cases, because credit was warranted for identification and corrective actions.

PENDING (EA 96-034): The staff is considering escalated enforcement action related to radwaste procedures.

PENDING (EA 96-059): The staff is considering escalated enforcement action for potential discrimination.

PENDING (EA 96-067): The staff is considering escalated enforcement action related to the inoperability of the standby gas treatment system.

PENDING (EA 96-086): The staff is considering escalated enforcement action related to the inoperability of the service water system.

PENDING (EA 96-106): The staff is considering escalated enforcement action related to the inoperability of the gas turbine emergency diesel generator due to an inoperable fuel pump.

PENDING (EA 96-145): The staff is considering escalated enforcement action related to potential violations involving design control, FSAR, and procedure problems related to the spent fuel pool and support systems.

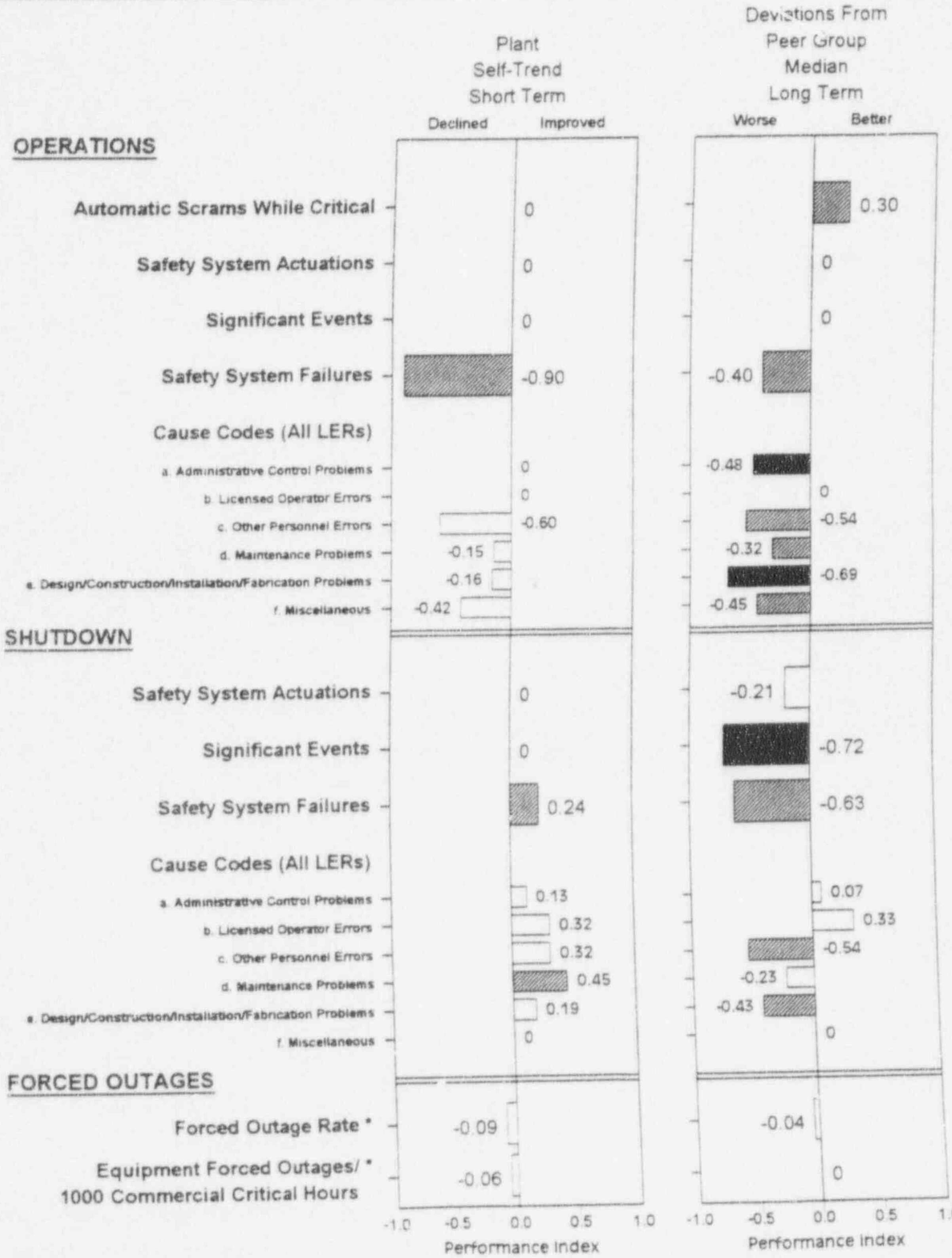
PENDING (EA 96-146): The staff is considering escalated enforcement action related to recirculation spray system (RSS) design problems that could result in the loss of service water to the RSS heat exchanger under certain LOCA conditions.

MILLSTONE 1

Peer Group: General Electric Pre-TMI

93-1 to 95-4 Trends and Deviations

Legend: Statistical Significance — High / Medium / Low

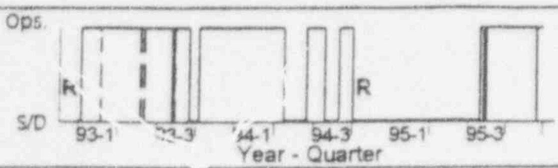


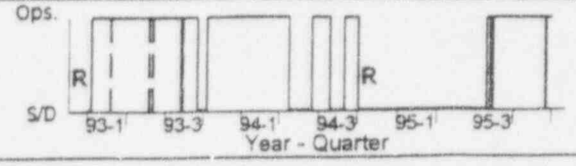
* Not Calculated for Operational Cycle

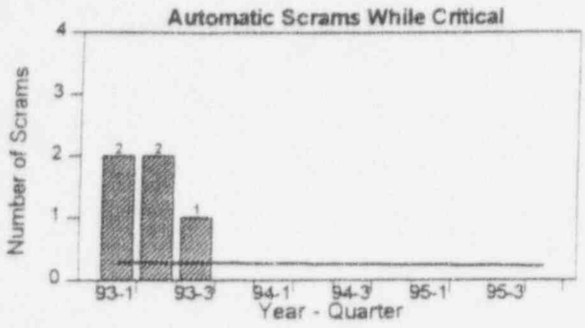
MILLSTONE 2

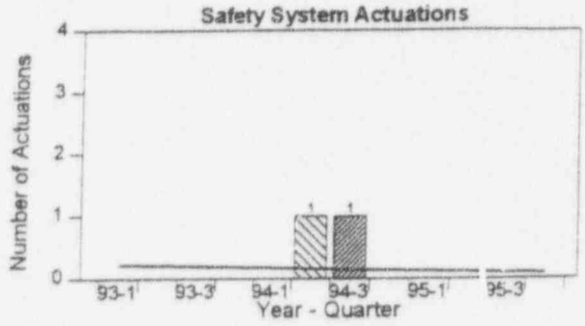
93-1 to 95-4 Quarterly Data

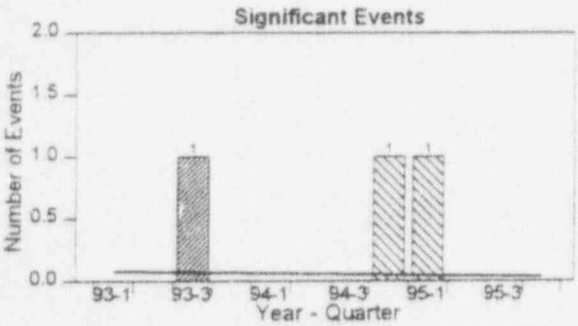
Legend:
Shutdown < approx. 72 hrs
Refueling R
Industry Avg. Trend
StartUp
Operation
ShutDown
Not Shown Using Op. Cycle

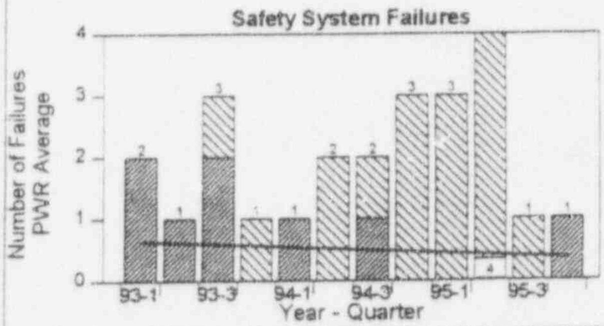


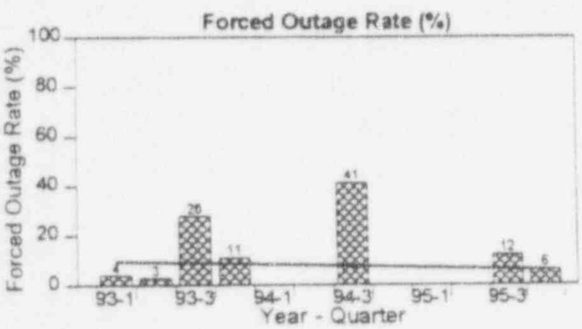


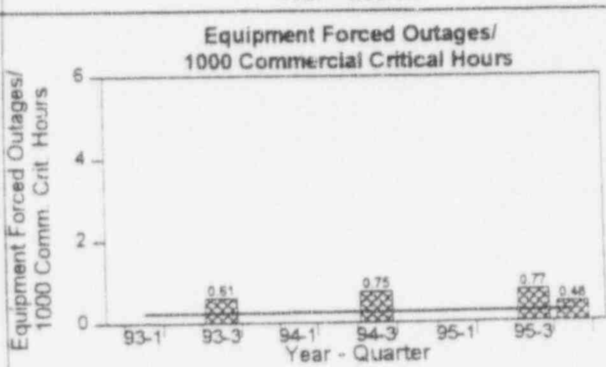


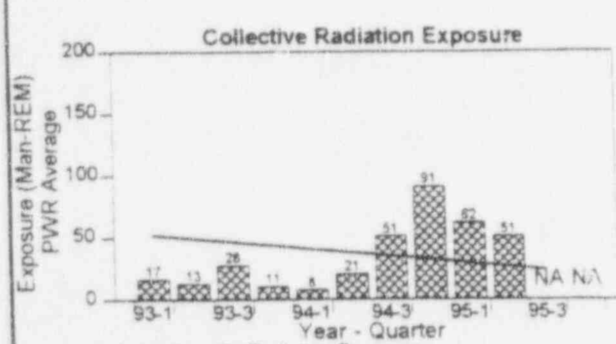




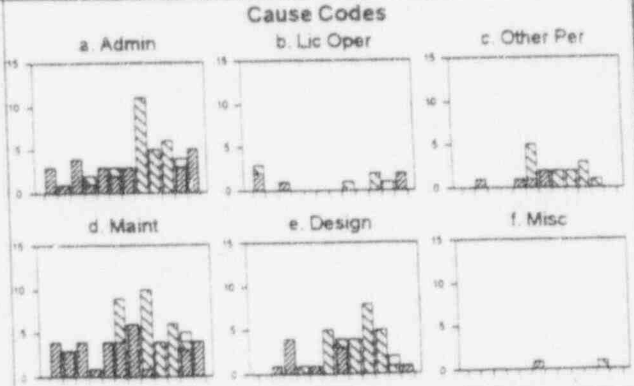








* Unit Specific Radiation Exposure



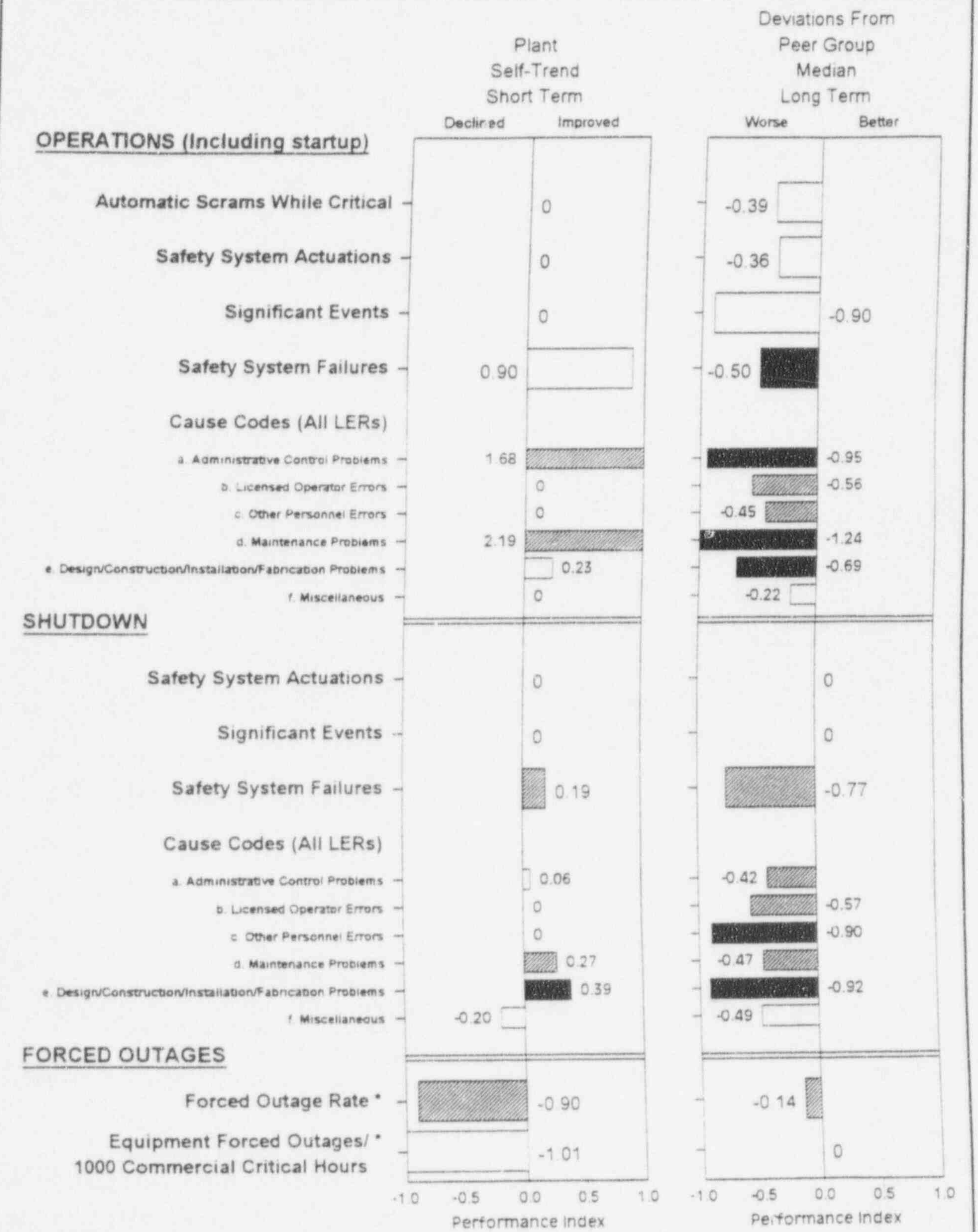
~~PREDECISIONAL~~

MILLSTONE 2

Peer Group: Combustion Engineering w/o CPC

93-1 to 95-4 Trends and Deviations

Legend: Statistical Significance — High / Medium / Low



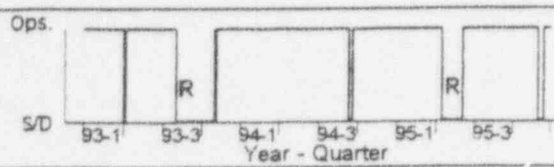
* Not Calculated for Operational Cycle

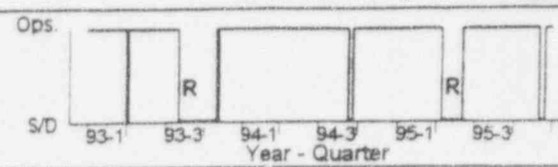
PREDECISIONAL

MILLSTONE 3

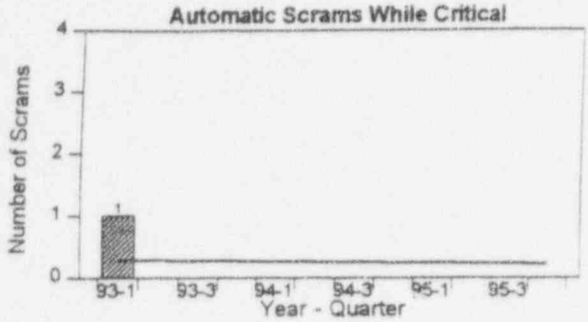
93-1 to 95-4 Quarterly Data

Legend:
Shutdown < approx. 72 hrs
Refueling R
Industry Avg. Trend
StartUp
Operation
ShutDown
Not Shown Using Op. Cycle

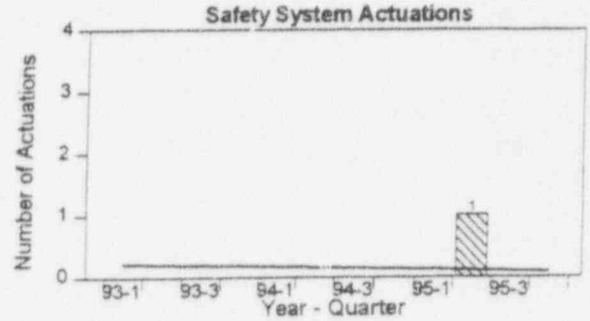




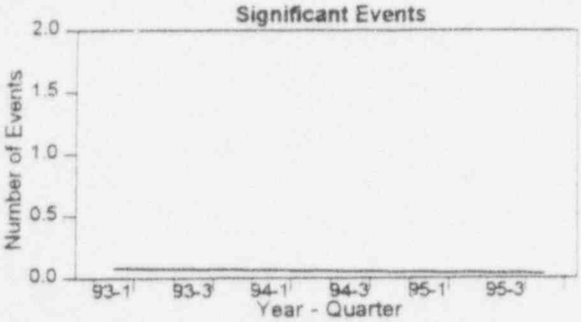
Automatic Scrams While Critical



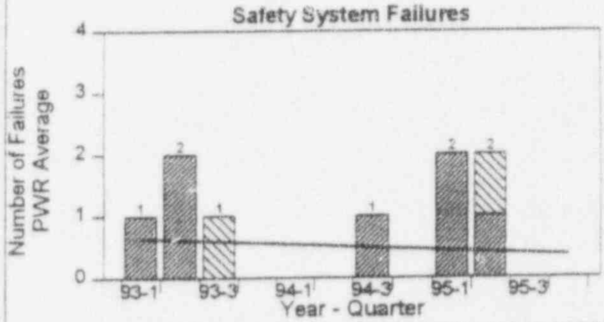
Safety System Actuations



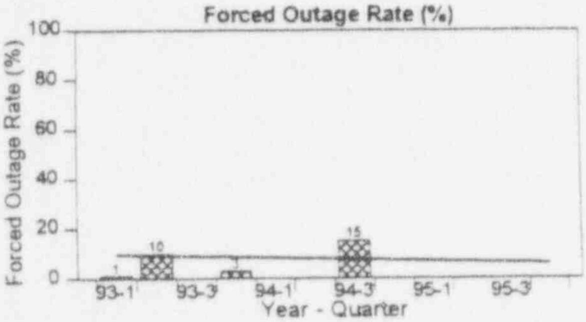
Significant Events



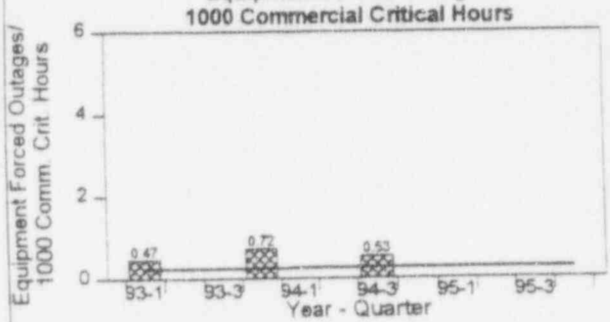
Safety System Failures



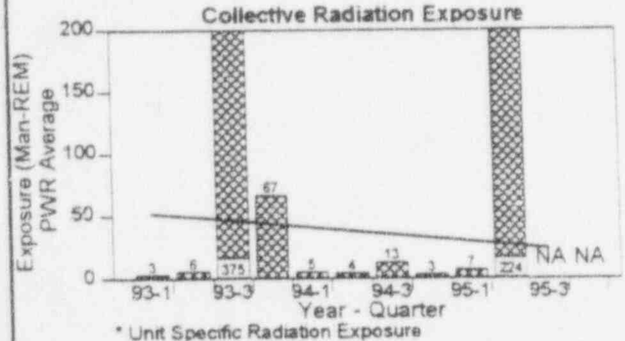
Forced Outage Rate (%)



Equipment Forced Outages/ 1000 Commercial Critical Hours

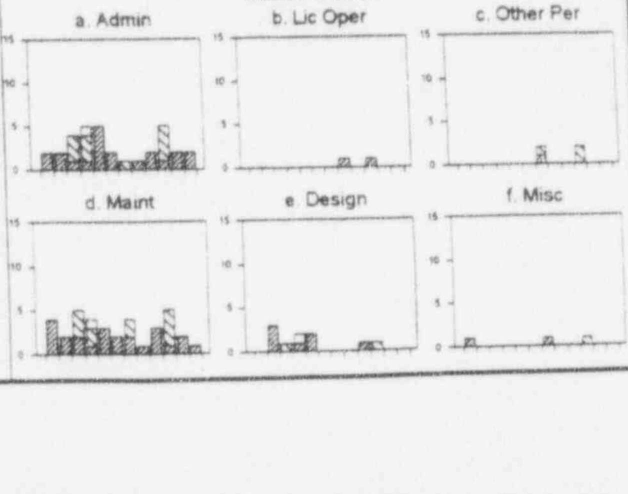


Collective Radiation Exposure



* Unit Specific Radiation Exposure

Cause Codes



~~PREDECISIONAL~~

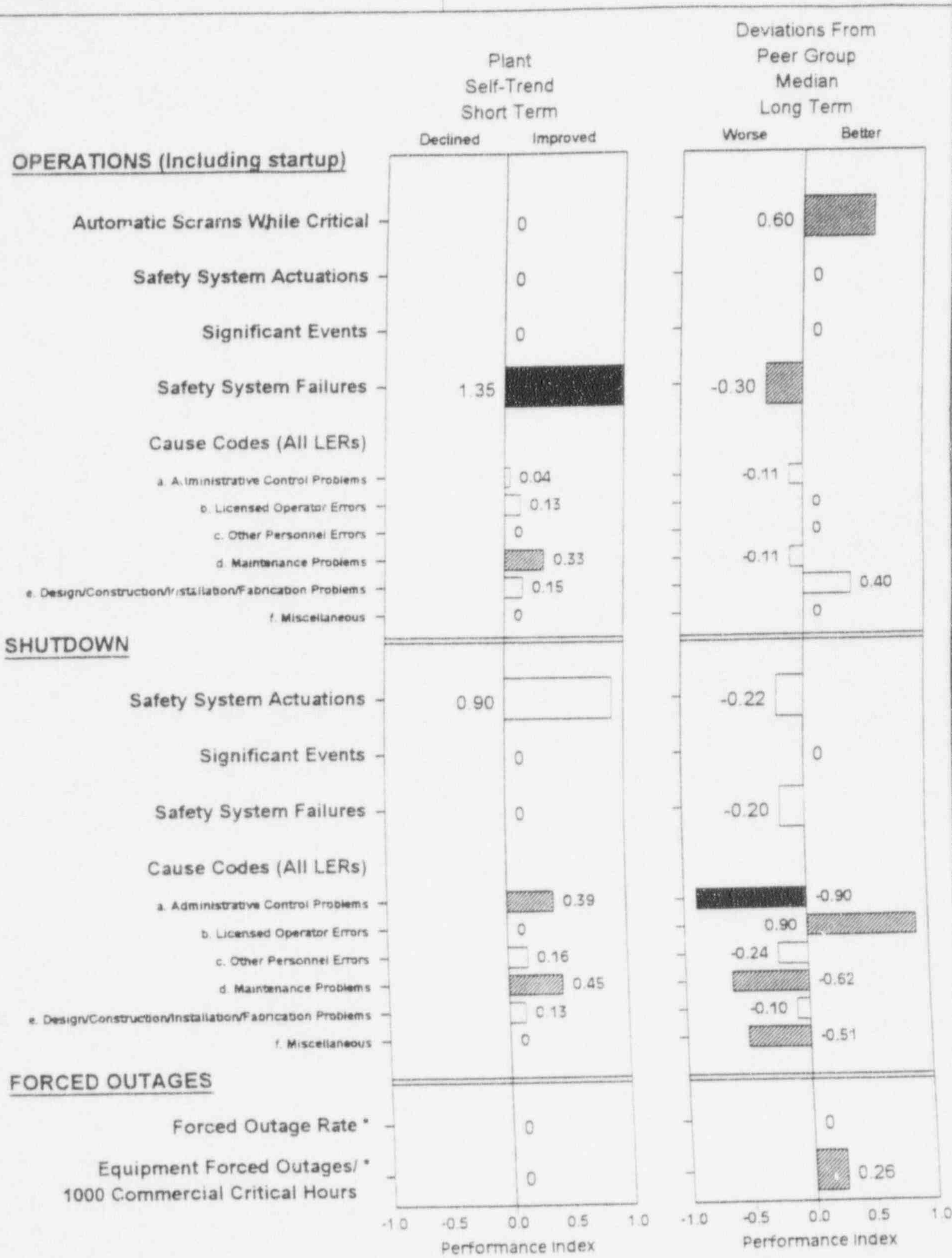
MILLSTONE 3

Peer Group:Westinghouse New 3 and 4-Loop

93-1 to 95-4 Trends and Deviations

Legend: Statistical Significance

High / Medium / Low



* Not Calculated for Operational Cycle

Biographical Data

Attachment 7

Northeast
Utilities System

Northeast Utilities Service Company
P.O. Box 270
Hartford, Connecticut 06141-0270

People Profile

BERNARD M. FOX

Bernard M. Fox is president and chief executive officer of the Northeast Utilities system (NU).*

He is also a trustee of NU and president and chief executive officer and a director of the Connecticut Yankee Atomic Power Company.

Born in New York City, ***Fox*** was graduated from Regis High School in 1959, Manhattan College in 1963 with a bachelor of science degree in electrical engineering, and in 1964 from Rensselaer Polytechnic Institute with a master's degree in the same discipline. In 1979, ***Fox*** completed the Program for Management Development at the Harvard Business School. In 1982, he completed a special program of study in auditing, legal affairs and financing activities, including a three-month assignment with Morgan Stanley and Company in New York City.

Fox joined The Hartford Electric Light Company (HELCO) in 1964 as a cadet engineer. In 1966, after HELCO had become a subsidiary of NU, he was named an engineer in NU's System Planning Department, and senior engineer in 1971. He transferred to the utility's Nuclear Engineering and Operations group in 1973. In 1979, he was appointed system director—Engineering Management Services, and system director—Transmission Engineering and Construction in 1980. He was appointed to the post of vice president and general manager—Gas in 1981 and was elected to the position of vice president and chief financial officer in 1983. In 1986 he was elected executive vice president and chief operating and financial officer. In 1987 he became president and chief operating and financial officer, and relinquished the position of chief financial officer in 1990. He assumed his present position in 1993.

Fox is chairman of the board of the Institute of Living. He also serves on the board of directors of Shawmut Bank Connecticut, N.A., Shawmut Bank, N.A., and Shawmut National Corp., Dexter Corporation, and the Connecticut Business and Industry Association. In addition, he is a member of The Mount Holyoke College BoardofTrustees, a member of the Board of the 1995 Special Olympics World Summer Games organization, Chairman of the Fidelco 1994 Walk-Run-Ride Event, and Ticket Sales Chairman of the 1994 Girl Scouts Woman of Merit Dinner.

Northeast Utilities System

Northeast Utilities Service Company
P.O. Box 270
Hartford, Connecticut 06141-0270

People Profile

ROBERT E. BUSCH

Robert E. Busch is president of the Energy Resources Group and chief financial officer for the Northeast Utilities system (NU).* He has overall responsibility for the company's nuclear and fossil/hydro operations, wholesale marketing, treasury, accounting, budget management, financial planning, and information resources.

Busch is a native of Cleveland, Ohio. He was graduated from Case Institute of Technology in 1968 with a bachelor of science degree in engineering. He earned a master of science degree in engineering from Rensselaer Polytechnic Institute in 1971 and a master of business administration degree, summa cum laude, from Northeastern University in 1981.

Prior to beginning his NU career, Busch worked on the Apollo Program and on research for the U.S. Navy in antisubmarine warfare. He joined NU in 1974 as a cost and schedule engineer and held various positions in the cost and scheduling area. In 1980 he became chief of cost and schedule control. In 1981 he became project manager of the Millstone Nuclear Power Station Unit 3 and brought the unit to completion. In 1986 he was named director—Special Financial Projects. In that capacity, he had a period of extensive training in corporate legal matters by Day, Berry & Howard, regulated utility accounting by Arthur Andersen & Co., and a four-month internship in public utility financing at Morgan Stanley & Company, Inc., of New York City, as well as completion of the Harvard Graduate School of Business Administration Program for Management Development. He was elected senior vice president in 1987 and, in 1990, was given the additional responsibility of chief financial officer. He was elected executive vice president in 1992, and he assumed his current position in January 1994.

He is a director of Connecticut Special Olympics. He is also a senior fellow of the American Leadership Forum, and a former member of the American Nuclear Society, from which he received a Certificate of Governance for his performance as chairman of the Connecticut chapter. He has received the Wall Street Journal Award for academic excellence in business education and has been elected to several honorary societies: Eta Kappa NU (for electrical engineering), Beta Gamma Sigma (for business) and Phi Kappa Phi (for academic performance).

[redacted]

* *NU is a registered holding company formed in 1966 whose principal operating-company subsidiaries are The Connecticut Light and Power Company, Holyoke Water Power Company, Public Service Company of New Hampshire, and Western Massachusetts Electric Company, and whose principal nonoperating subsidiaries are North Atlantic Energy Corporation, North Atlantic Energy Service Corporation, Northeast Nuclear Energy Company, and Northeast Utilities Service Company. In addition, Charter Oak Energy, Inc., and HEC Inc., are NU's nonutility subsidiaries.*

January 1994

Volume 1 ◆ Number 2 ◆ January 18, 1996

Millstone Today

A Daily Publication for Millstone Units 1, 2, and 3

Meet Ted Feigenbaum

Starting February 1, Ted C. Feigenbaum becomes executive vice president and chief nuclear officer for NU's nuclear program. Ted will have overall responsibility for NU's nuclear program as it goes through the upcoming transition and enters the arena of deregulation.

Ted C. Feigenbaum

Asked what his priorities will be as he begins his tenure, Ted responded:

"It's hard to capsulize my priorities in one or two statements, because we have so many challenges ahead. But we will be better equipped to face all those challenges if we establish two things:

"--First, a passion for safety. Not just a safety ethic, and not just the statement that safety comes first, we need to establish that, without safety, we will ultimately lose, regardless of what challenges we face. This passion has got to permeate the entire organization, every level.

"--Second, I want to reconnect our employees with the Company. We cannot win unless everyone is pulling in the same direction, and we can't do that until we get better at listening to our employees. It all boils down to respect: If we respect each other as coworkers and individuals, we can establish common goals. I know that's a long row to hoe, but if we are successful, there is nothing we can't do as an organization.

"As is always the case, words without actions ring hollow. I hope to demonstrate through my actions that I am serious about these two points. And I hope to spend as much time as I can early this year visiting different areas of the organization, getting to know people, and listening to their ideas and concerns."

Ted was born and grew up in New York City. He still sports vestiges of a New York accent, and still loves to quote Yogi Berra, but if you talk about the Yankees, he'll admit that he has been indoctrinated into the New England sports scene and is now a Red Sox fan.

Ted received a bachelor of engineering

(Continued on reverse)

degree in mechanical engineering from the City College of New York in 1972 and is a registered professional engineer. In 1994, he completed the Advanced Management Program at Harvard University Graduate School of Business Administration.

Ted has been in the power plant business since 1972, as a lead mechanical engineer for Stone & Webster, and then as project engineer at St. Lucie in Florida. He joined the Seabrook project in 1984, and has held several management positions at Seabrook since that time.

He became senior vice president and chief operating officer in 1989, and was elected president and chief executive officer of New Hampshire Yankee in 1990.

He assumed his current position, chief nuclear officer for North Atlantic Energy Service Corporation in 1992.

[redacted]

"I look forward to working with all my new colleagues in Connecticut," said Ted.

Organization Charts

Attachment 8

NORTHEAST NUCLEAR ENERGY COMPANY
MILLSTONE UNIT 1, 2, & 3
SEABROOK, HADDAM NECK

Appendix D: June 1995 SMM Narrative Summary

PRE-DECISIONAL

MILLSTONE

I. HISTORY

This is the eighth SMM during which Millstone has been a full discussion plant. The station was not discussed during the SMM in June 1993 due to indications of improved performance, but was discussed at the four prior meetings and the three subsequent meetings. This station has never been placed on the Watch List.

During the previous SMMs, information was provided that detailed the continuing performance problems at Millstone Station, Northeast Utilities' (NU's) Performance Enhancement Program (PEP), recent escalated enforcement actions, and the diminished frequency of allegation receipts. In early 1992, in response to an overall decline in performance and with the assistance of management consultants, NU developed the PEP to ensure the effective use of resources, and to implement the recommendations of four internal performance assessment task forces. The PEP was a long-term effort (five-years) that provided substantial additional staff and budgetary resources to the nuclear organization and detailed a broad range of corrective action plans to address a number of performance concerns. To monitor the PEP implementation, the NRC commissioned the Millstone Assessment Panel (MAP) in June 1992. The effectiveness of the PEP in achieving performance improvement was somewhat limited, as evidenced in plant operational performance, particularly at Unit 2. Due to the engineering integration effort and budgetary constraints, staffing increases originally included in the PEP were not reached (i.e, about 310 additional positions rather than 450). In October 1994, the PEP was absorbed into the overall NU "Business Plan."

Prior to 1992, the frequency of allegations received from Millstone employees was quite high. Principally, employee concerns were relayed to the NRC by three individuals. The focus of these allegations was procedure adherence and work control, as well as harassment, intimidation and discrimination (HI&D). In November 1991, two of the three were fired, allegedly for irreconcilable differences with the licensee. Subsequently, the rate of allegation receipt diminished abruptly until July 1993 when the rate again increased to over four concerns per month (a number of which contained multiple subparts), exclusive of a number of concerns raised via the 10 CFR 2.206 process. The focus of recent allegations has been work control and management's handling of employee concerns at Millstone Unit 1. The third employee previously noted reached a settlement with NU in early 1993 and resigned. The harassment, intimidation and discrimination of this employee during 1989-1990 by licensee management was the subject of a $100,000 civil penalty issued to NU on May 4, 1993. On March 17, 1994, NU was informed that the staff concluded there was insufficient evidence of HI&D involving the termination of the other two employees noted above for whistleblowing activities. In July 1994, the NRC issued two Severity Level II violations to NU, one for their failure to take prompt corrective action concerning the operability of the Unit 1 feedwater coolant injection system (FWCI) and the other involving discrimination against the engineer who raised these operability concerns. In early December 1994, NU made a presentation to the NRC detailing their most recent efforts to handle employee concerns, most notably an increased focus on training and counseling supervisors on how to better handle employee concerns. Since that time, the NRC has noted a reduction

K/1

3 Operations Manager to become the Unit 2 Operations Manager and the temporary assignment of select managers to support the Operations Manager during their restart recovery efforts. Senior NU management has also employed the use of four organizational consultants since mid-1994 to help elicit employee involvement in improving performance as well as assessing and helping to improve the organizational culture. The results of their work are not yet evident as much of their efforts are still in progress.

III. FUTURE ACTIVITIES

Five resident inspectors are currently assigned to the site (1 SRI, 4 RIs). Two region-based Project Engineers are currently devoted full-time to Millstone and Haddam Neck to provide timely follow-up and management of allegations and to support the Millstone inspection program.

A RATI is scheduled to be conducted at Millstone Unit 2 prior to restart from the current refueling outage. As of this writing, the RATI is scheduled to start the week of May 30, 1995. The team will assess their readiness for restart and the progress of their performance improvement efforts. Other significant inspection activities include a review of the Millstone procurement program as well as a follow-up of NRC concerns with the NSCP and the resolution of multiple employee concerns in the Unit 1 Maintenance area which were referred to NU for their resolution. Region I is also currently negotiating with NRR for support in closing the extensive number of open inspection items generated at Millstone Station associated with their marginal performance.

The next refueling outage for Unit 1 is scheduled for October 1995 while the ongoing refueling outage at Unit 2 is nearing completion. Unit 3 commenced a scheduled 53 day refueling outage on April 14, 1995.

DATA SUMMARY

I. OPERATIONAL PERFORMANCE

A. Scram Summary

Unit 1 - None.

Unit 2 - None.

Unit 3 - None.

B. Significant Operator Errors

Unit 1

No noteworthy operator errors occurred since the last SMM.

Unit 2

On February 7, 1995, operators failed to vent the "C" HPSI pump prior to operation, resulting in damage to the pump. In addition, following repairs to the pump, tagging controls were not properly implemented on the pump casing vent, resulting in the spillage of about 20 gallons of contaminated water through the vent.

Unit 3

No noteworthy operator errors occurred since the last SMM.

C. Procedures

Improvement in the use and adequacy of Operations and Instrumentation and Control procedures and in training were noted at Units 1 and 3. Unit 3 was also found to have made improvements in the adequacy and use of procedures in other plant departments. However, lapses did occur in procedure adherence and compliance with Technical Specifications, principally at Unit 2, where longstanding problems with operator attention to detail and procedural adherence were evident. Unit 2 Operations department management's tolerance of existing deficient conditions and/or inability to effect positive change contributed to operators willingness to work around less-than-adequate procedures and utilize informal configuration control. Significant effort has recently been devoted toward improving performance in this area at Unit 2 prior to restart.

An NU contractor recently identified a number of deficiencies in the Unit 2 Emergency Operating Procedures (EOPs) during an independent review of their adequacy. The scope of the EOP deficiencies is not yet clear, although the resolution of these deficiencies could

likely impact Unit 2 restart.

II. CONTROL ROOM STAFFING

A. Number of Licensed Operators

	SRO	RO	LSRO	TOTAL
Unit 1	29	16	0	45
Unit 2	29	17	0	46
Unit 3	27	16	0	43

B. Number and Length of Shifts

Six, 8 hour shift for each unit.

C. Role of STA

All three Millstone units use dual role SROs/STAs. However, efforts are in progress to provide a separate STA on each shift.

D. Requalification Program Evaluation

In October 1994, the NRC conducted a requalification program evaluation at Millstone Unit 1. The program was determined to be effective and operator performance was considered satisfactory.

In June 1994, the NRC conducted a requalification program evaluation at Millstone Unit 2. The program was determined to be effective and operator performance was considered to be satisfactory following resolution of two program concerns: (1) non-challenging written examinations, and (2) facility evaluation of operator performance during the simulator scenarios did not match the NRC evaluation. As a result of these concerns, a followup examination was conducted. Two additional crews were evaluated and demonstrated satisfactory performance on a dynamic simulator operating test. Each crew was administered one NRC-developed scenario and one facility-developed scenario.

In September 1994, the NRC conducted a requalification program evaluation at Millstone Unit 3. The program was determined to be effective and operator performance was considered to be satisfactory. The program continues to show improvement when compared with previous years.

III. PLANT-SPECIFIC DATA AND UNIQUE DESIGN INFORMATION

A. Plant-Specific Information

Plant	Millstone 1	Millstone 2	Millstone 3
Owner	NNECO	NNECO	NNECO
Reactor Type	BWR-3 GE	2-Loop CE PWR	4-Loop W PWR
Capacity, MWe	660	860	1150
AE/Constr.	Ebasco	Bechtel	Stone & Webster
CP Issued	5/19/66	12/11/70	8/9/74
OL Issued	10/31/86	9/30/75	1/31/86
Comm. Oper.	3/1/71	12/26/75	4/23/86

B. Unique Design Information

Unit 1

Containment: BWR 3 with Mark I containment (hardened vent added during the cycle 14 refueling outage)

ECCS: Feedwater coolant injection (FWCI) system provides high pressure injection for small-break LOCA (no HPCI or RCIC); isolation condenser provides cooling when reactor is isolated from the main condenser; capacity for 105% bypass of steam to the main condenser.

AC Power: Unit 1 has three sources of offsite power to the plant transformers and the capability to receive power from a Unit 2 diesel generator to a Unit 1 4160 Vac safety bus. One emergency diesel generator (2.7 MWe) and one gas turbine (11.5 MWe) generator provide normal emergency AC power.

DC Power: The unit has an 8 hour station blackout rating.

Unit 2

Containment: Dry, prestressed concrete with steel liner.

Steam Generators: Replaced both steam generators in 1992 due to previous chronic and excessive tube leakage problems.

ECCS: Four safety injection tanks; three high pressure injection pumps (also used as charging pumps); two low pressure injection pumps.

AC Power: Offsite AC power supplied via four 345 kV lines, the main and reserve station service transformers and associated power buses; two independent diesel generators and two radial distribution systems.

DC Power: Two vital DC distribution systems with two spare battery chargers (full capacity).

Unit 3

Containment System: Subatmospheric (operated at about 1 psi below atmospheric versus 5 psi as originally designed) dry containment with steel lined reinforced concrete and an enclosure building.

ECCS: Four accumulator tanks; three charging pumps; two safety injection pumps; and two RHR pumps.

AC Power: Two independent 345 kV offsite power lines; two 100% capacity diesel generators (4986 kWe), one SBO diesel (2260 kWe).

DC Power: Four Class 1E batteries.

Shared Systems

Heat Sink: Long Island Sound.

Manual cross-tie of Unit 2 diesel generator to Unit 1 emergency bus.

Station Blackout: The Millstone site has a minimum blackout coping duration. Unit 3 installed an additional air-cooled diesel generator for alternate AC power; Units 1 and 2 have adequate alternate AC power sources to meet SBO requirements.

IV. SIGNIFICANT MPAS OR PLANT-UNIQUE ISSUES

Unit 1

MPA F-71 - Detailed Control Room Design Review: The SER was issued 7/10/90. The modifications are scheduled to be completed during the 1995 refueling outage.

Unit 2

MPA B-111 - Individual Plant Examination: NU's submittal is currently being reviewed with a scheduled completion date of July 1995.

Unit 3

MPA B-118 - Individual Plant Examination of External Events: NU's submittal is currently being reviewed with a scheduled completion date of July 1995.

Reduce Minimum Reactor Core Flow: NU's submittal is currently being reviewed.

V. STATUS OF THE PHYSICAL PLANT

A. Problems Attributed to Aging

None.

B. Other Hardware Issues

Unit 1

During the 1994 refueling outage, General Electric performed a visual inspection of the core shroud using remotely operated closed circuit video equipment. The inspection revealed several small cracks on both the inner and outer side of the shroud. After analyzing the results, the licensee concluded that the indications were well below the screening criteria prepared by GE and that they do not pose a concern for plant operation. NU plans to inspect the core shroud again during the 1995 refueling outage. NU is also following industry technical concerns with BWR core shroud cracking.

Millstone Unit 1 has a history of safety relief valves (SRVs) failing to meet the Technical Specification "as-found" tolerance limits. The cause of the setpoint "drift" has been attributed to oxide bonding of the seat and disk in the pilot valve of the SRV. At the recommendation of the BWR Owners Group SRV Setpoint Drift Subcommittee, NU replaced half of the pilot valves with valves having a new disk material - a platinum stellite alloy. Operating experience with this new material is still being evaluated, but has not been promising to date. NU is also considering adding pressure switches to the valves to utilize the solenoid-assist mode of the SRVs. The NRC staff is reviewing this generic industry issue.

Units 1 and 2

Significant hardware issues at Millstone station include the calculated stress levels experienced by some MOVs at Units 1 and 2 under certain transient scenarios, an issue that has received close NRC scrutiny.

Unit 2

During the current refueling outage, Unit 2 determined that the two emergency core cooling system (ECCS) sump valves were susceptible to pressure locking. Modifications to the valves have been completed, and mock-up testing of the valves to determine their actual susceptibility to pressure locking is ongoing. Unit 2 is also continuing to follow previously identified degradation in spent fuel pool Boraflex poison material as well as making various material condition improvements to the plant to improve appearance and remedy problems identified during the current outage.

Unit 3

Millstone Unit 3 began a refueling outage on April 14, 1995, which is scheduled to end in early June. Major activities scheduled for the outage include: completing the GL 89-10 MOV testing program, service water erosion/corrosion inspections and replacements, and steam generator eddy current testing for two steam generators (the other two were inspected in 1993).

Site

Chronic service water leaks require frequent attention. NU is replacing piping, particularly at Unit 2 most recently, using different material in some cases and is studying long-term corrective actions.

VI. PRA

A. PRA Insights

Unit 2

Millstone 2 is an early Combustion Engineering NSSS plant. If all feedwater is lost at a PWR, the plant must depend on bleed and feed to prevent core damage. The two turbine-driven main feedwater pumps have low suction pressure trips. The auxiliary feedwater system (AFW) has good redundancy (two motor driven AFW pumps and one turbine driven pump). However, should all feedwater be lost, the bleed and feed capability at this unit is minimal. The Safety Injection (SI) pumps (used for high pressure injection) have a shutoff head of 1205 psig. The three positive displacement charging pumps, which would have to be aligned manually, have low capacities (44 gpm each) and provide little feed and bleed assistance. Since the two PORVs are small, operator action probably must be very rapid to assure success in the event that bleed and feed is needed.

Station blackout is considered to be an important contributor to estimated core damage frequency for many plants. As reported in the FSAR, the station batteries have only a one hour capacity. The alternate AC power source is the crosstie to either one of the Unit 1 emergency AC power sources. The alternate AC source can be made available within 1 hour from the onset of an SBO. During refueling outage 11, which ended in Jan. 1993, weather protection was installed to the alternate AC components in order for Unit 2 to fully comply with the SBO rule. The Millstone site has been subject to several extended losses of offsite power due to hurricanes. The licensee has taken the position that it will shut down the reactor if a hurricane is in proximity.

Service water/component cooling system failure is considered to be a potentially significant contributor to core damage frequency at many

plants. At Unit 2, in the event of an accident, non-safety Reactor Building Closed Cooling Water system loads must be isolated from the safety loads. Also, temperature control valves on the associated heat exchangers must go to the full open position to assure system success. The Unit 2 service water system piping has experienced significant corrosion/erosion. The licensee has replaced most of the large piping with plastic lined piping and is continuing its ongoing surveillance program and replacement of piping, as necessary.

The licensee uses PRA insights in reviews of conceptual design changes. To a lesser extent, the PRA is used for support of engineering and operations. Millstone 2 is considering joining Millstone 1 and Haddam Neck in the ISAP program.

B. PRA Profile

A Level 1 PRA was completed by Millstone 2 in the fall of 1991. The licensee submitted their IPE, which consisted of an updated Level 1 PRA coupled with Level 2 PRA, dated December 30, 1993, in response to GL 88-20. The IPE review commenced on November 8, 1994, and is expected to be completed in June 1995. The IPE reported that the core damage frequency (CDF) due to internally initiated events is 3.4E-5/yr. The contributions to CDF by initiating event group are listed below:

Initiating Event Group	CDF/yr	% of Total CDF
Loss of Offsite Power	8.4E-06	24.6
Transients	6.1E-06	17.8
Loss of DC Bus A	3.9E-06	11.4
Loss of DC Bus B	3.9E-06	11.4
Steamline Break	2.9E-06	8.5
Large LOCA	1.7E-06	4.8
Small LOCA	1.6E-06	4.8
Small-small LOCA	1.5E-06	4.3
Medium LOCA	1.3E-06	3.7
Loss of Service Water	9.7E-07	2.8
Loss of Vital AC Panels 10&30	8.4E-07	2.5
SGTR	5.2E-07	1.5
Main Feedline Break	2.4E-07	0.7
Internal Flooding	2.0E-07	0.6
Intersystem LOCA	6.6E-08	0.2

The results from the original Level 1 PRA led to several plant changes designed to decrease the overall CDF, and these changes have been reflected in the updated Level 1 PRA. Some of the changes resulted in a decrease of AFW unavailability by increasing the ability to isolate an intact SG from a faulted SG. Another change required additional testing of LPSI check valves for reverse flow, thereby increasing their availability and reducing the possibility of failing LPSI cold leg piping due to overpressurization.

Based on the screening criteria in the IPE, no major vulnerabilities were identified. However, a potential vulnerability sequence had been identified. The sequence involves reactor coolant pump thermal barrier failure leading to possible overpressurization of reactor building closed cooling water and a small intersystem LOCA. This sequence is outside the design basis of the plant. The licensee is evaluating the impact of this sequence on the mean core damage frequency. The licensee is also planning to take actions to address the potential vulnerability sequence before the next refueling outage.

The Level 2 analysis found that the conditional probability of early containment failure is approximately 9.7 percent, and is dominated by direct containment heating. The conditional probability of late containment failure is approximately 33.1 percent, and dominated by the basemat melt-through failure. The conditional probability that the containment remains intact is 57.2 percent.

The licensee has indicated that the IPEEE will be submitted in December 1995.

C. Core Damage Precursor Events

On the basis of the precursors identified by ORNL for 1993 (NUREG/CR-4674, vols. 19 and 20) and the preliminary precursors for 1994, the staff did not identify any precursor events for the site that have a conditional core damage probability of 1E-5 per year or greater.

The following events have been classified as "Significant Events" for the Performance Indicator Program:

On August 5, 1993, an unisolable 5 gpm RCS leak developed while the unit was at 100% power. Repeated applications of leak sealant caused one of four body-to-bonnet studs to shear on an unisolatable manual letdown isolation valve which created the 5 gpm RCS leak. If the other three studs had sheared, a non-isolable small LOCA would have ensued. The Conditional Core Damage Probability for the event was preliminarily estimated to be [7.1E-4] X [probability of inducing a small LOCA after the first bolt sheared]. Long term cooling concerns of the potential small LOCA were raised since the centerline of the valve is 8 to 10 inches below the centerline of the cold leg. If the small LOCA had occurred, HPSI would have had to continue while the plant entered and maintained RHR mode until the break was fixed.

On December 6, 1994, the licensee identified a release path from the Enclosure Building that would allow a direct discharge to atmosphere bypassing the charcoal filters following a LOCA. Specifically, a hydrogen analyzer cabinet and sample hood exhaust fan were found to take a suction on the enclosure building and discharge out the Unit 2 Main Exhaust stack. This flow path had HEPA filters but lacked charcoal filters. Given this release path, the licensee calculated that the site

boundary thyroid dose would be more than twice the 10 CFR 100.11 limit in the event of a design basis accident, core melt, and subsequent fission product release.

On January 26, 1995, Millstone Unit 2 determined that both containment sump recirculation gate valves may experience pressure locking during a design-basis LOCA and fail in the closed position. A recent assessment by the licensee indicated that these valves would have been able to perform their intended function; however, the staff has not been able to verify this. The failure of these valves would make the water source for the ECCS and the containment spray unavailable during the recirculation phase of the LOCA. If these valves had been inoperable at times in the past, then the estimated increase in core damage frequency would be in the E-3/yr range, assuming that the recirculation function is not recoverable and that it is essential for mitigation of medium and large LOCAs. If recovery is possible, the estimate would be reduced. Opportunities for recovery might include manually opening a containment sump valve, or obtaining additional borated water from Millstone Unit 3. Based on the licensee's recent determination that the valves were operable, the NRC Events Assessment Panel re-examined this event and kept it as a "Significant Event" based on programmatic weaknesses in the licensee's management of the plant.

VII. ENFORCEMENT HISTORY

5/93 CIVIL PENALTY — The action was taken based on one Severity Level II violation related to the harassment, intimidation and discrimination by management above first-line supervisors against an corporate instrumentation and control engineering supervisor for raising safety issues. A Severity Level III violation was also noted related to the destruction of an original Safety Evaluation and modification of a document in order to conceal safety concerns. A civil penalty was issued for the Severity Level II violation to emphasize the importance of licensees providing a work environment that is free of harassment, intimidation and discrimination against those who raise safety issues. A civil penalty was not issued for the Severity Level III violation because the licensee reported the matter and a "Director or Responsible Officer," as defined in 10 CFR Part 21, was not involved in the violation. The civil penalty for the Severity Level II violation was escalated because of significant management involvement in the violation. ($100,000)

8/93 ENFORCEMENT CONFERENCE — The staff considered enforcement action at Unit 2 involving: (1) partial loss of normal electrical power and subsequent Spent Fuel Pool drain-down which occurred July 6, 1992, and (2) the possible failure to take prompt action to correct single failure design discrepancies identified by the licensee. The staff concluded that escalated action for failure

to take prompt corrective action for the Spent Fuel Pool drain-down was not warranted. In addition, the staff decided not to cite violation for the design discrepancies due to the age of the violation (the original modification occurred in 1978) and the licensee's corrective actions.

9/93 CIVIL PENALTY — The action was based on a Severity Level III problem consisting of two violations. The violations involved: (1) the failure of numerous licensed operators in Units 1 and 2, to complete the licensed operator requalification training (LORT) program for the 1991 and 1992 requalification training period that ended December 31, 1992, and (2) the failure of the facility Nuclear Review Board (NRB), for the last six years, to either perform, or perform adequately, audits of training, retraining, qualification, and performance of the operations staff in Units 2 and 3 as required by the facility Technical Specifications and the Quality Assurance Plan commitment to ANSI 18.7-1976. A civil penalty was issued to emphasize the importance of adequate and continuing management attention to the LORT, so as to assure all training requirements are completed in a timely manner, and appropriate audits are performed to verify completion. The civil penalty was escalated because of NRC identification of the deficiencies but mitigated based on the licensee's prompt and comprehensive corrective actions. ($50,000)

12/93 CIVIL PENALTY — The action was based on three violations at Unit 2 that, in the aggregate, were classified as one Severity Level III problem. The first violation concerned the licensee failing to perform a safety evaluation, in accordance with the requirements of 10 CFR 50.59, prior to performing a leak repair of an unisolable letdown line valve. The second violation of this action consisted of 10 examples of the licensee failing to provide workers adequate and appropriate instructions and license personnel failing to follow existing procedures. All of these 10 examples were associated with the attempted leak repair of the letdown line valve. The third violation identified the licensee's failure to perform adequate inspection of activities affecting quality, and this violation also stemmed from the licensee's attempt to repair the letdown line valve. The civil penalty that was issued for this action was escalated due to the problems being identified by the NRC, weak corrective actions taken by the licensee, poor past performance, and the extended duration of the problem. ($237,500)

3/94 CIVIL PENALTY — The action was based on two violations at Unit 3 that constituted one Severity Level III problem. The first violation concerned design deficiencies in the auxiliary building filter system (ABFS) and the ability of that system to maintain negative pressure inside the secondary containment

following a postulated accident. Under certain conditions, this design deficiency caused the ABFS to be inoperable and this resulted in a violation of the requirements of the technical specifications. The second violation consisted of the licensee failing to take adequate measures to ensure that the adverse condition that existed with the ABFS was identified and corrected in a manner that would preclude recurrence. The civil penalty for this case was mitigated due to the licensee identifying the problem and taking prompt and comprehensive corrective action, and escalated because of poor past performance such that, on balance, no adjustments were made to the base civil penalty. ($50,000)

6/94 LETTER TO LICENSEE - The action was based on an investigation resulting in an OI conclusion that a licensee official had deliberately failed to provide complete and accurate information regarding a harassment and intimidation issue. Although the staff concluded that no regulatory action was warranted, the letter was to emphasize to the licensee the importance of submitting information that is complete and accurate in all material respects.

7/94 CIVIL PENALTIES AND DEMAND FOR INFORMATION — The action was based on an investigation by the Office of Investigations and an inspection conducted by Region I, concerning two violations at Unit 1 consisting of: (1) the deliberate delay in taking corrective actions for a condition adverse to quality (specifically a deliberate delay in the determination of the operability of the feedwater coolant injection system) that continued from June 1989 to November 1989, and (2) discrimination by an unit Engineering Manager against an engineer who was involved in the operability determination. Civil penalties were issued to emphasize the importance of prompt resolution of potential safety concerns when they exist, as well as ensuring that appropriate controls exist to preclude discrimination of employees who raise such concerns. In addition, a Demand For Information was issued to the licensee demanding an explanation why the NRC should not issue an Order to modify their license to preclude the Engineering Manager from any involvement in future licensed activities at any of the licensee's facility. For the first violation, the civil penalty was escalated to $120,000 due to the continuing nature of the violation when spanned a number of months in 1989. ($220,000)

8/94 CIVIL PENALTY — This action was based on two violations at Unit 2. The first violation consisted of three examples of failure to classify and notify NRC of events at Unit 2 that constituted Unusual Events under the licensee's Emergency Plan and Procedures. The second violation involved the failure to ensure that adequate shutdown margin requirements existed following the

identification of an immovable control rod assembly. These violations have been categorized in the aggregate as a Severity Level III problem. To emphasize the importance of appropriate classification and notification to the NRC of an Unusual Event and maintaining adequate shutdown margin to protect health and safety of the public, a civil penalty was issued after escalation of 50% for both identification and licensee performance and mitigation of 25% for the licensee's corrective actions. Enforcement discretion was exercised for a third violation that involved the compromise, since 1987, of the automatic primary containment isolation function for a postulated reactor water cleanup system line break. Discretion was exercised because the problem originated more than seven years ago, it was not a willful violation, adequate corrective actions have been taken by the licensee, and it was licensee-identified. ($87,500)

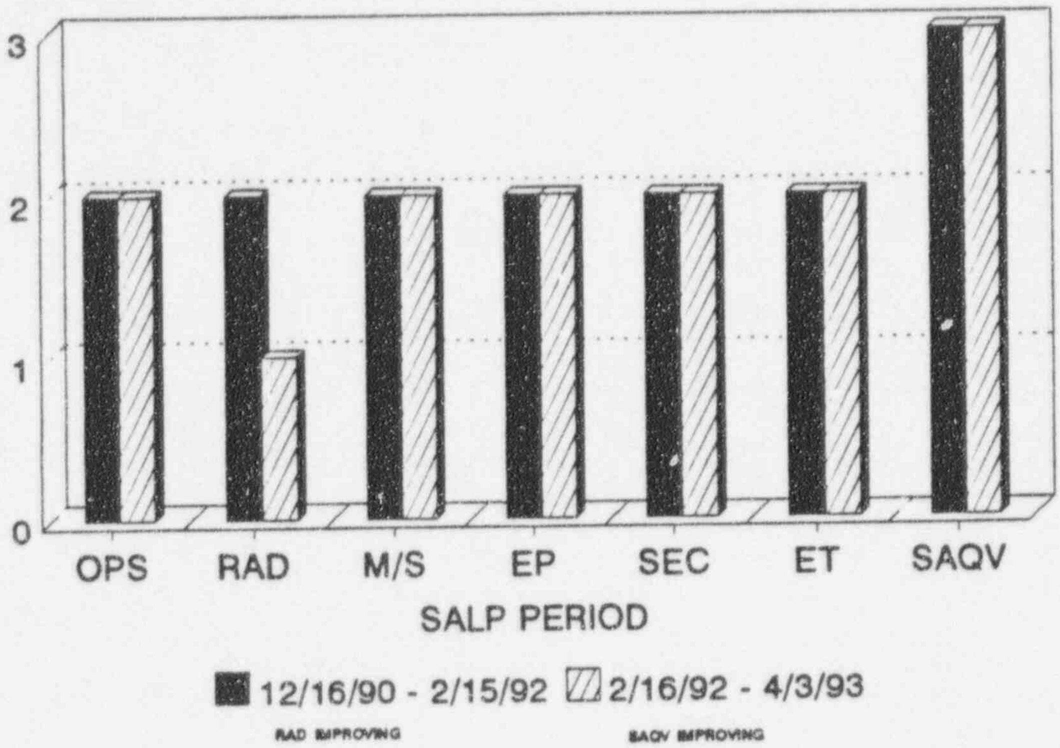
2/95 LETTER OF REPRIMAND - Based on an Office of Investigation Report dated August 5, 1994, the staff issued a letter of reprimand for apparent falsification of training records by a former supervisor of Operator Training at Unit 1.

2/95 ENFORCEMENT DISCRETION - This action was based on the licensee's identification of an oversight in the original design of the discharge flow path for the hydrogen analyzer ventilation system which could have permitted excessive offsite doses in the event of a loss of coolant accident. Enforcement discretion was exercised for the apparent Severity Level III violation because: (1) the condition was identified by the licensee's staff as a result of a healthy questioning attitude and was promptly reported to the NRC; (2) the condition existed since initial startup, was difficult to discover, and such identification was not likely by routine inspection, surveillance, and quality assurance activities; (3) comprehensive corrective actions were taken within a reasonable time period that involved an adequate root cause determination and a review for failures caused by similar root causes; and (4) the condition was caused by an old performance failure that is not reasonably linked to present day performance.

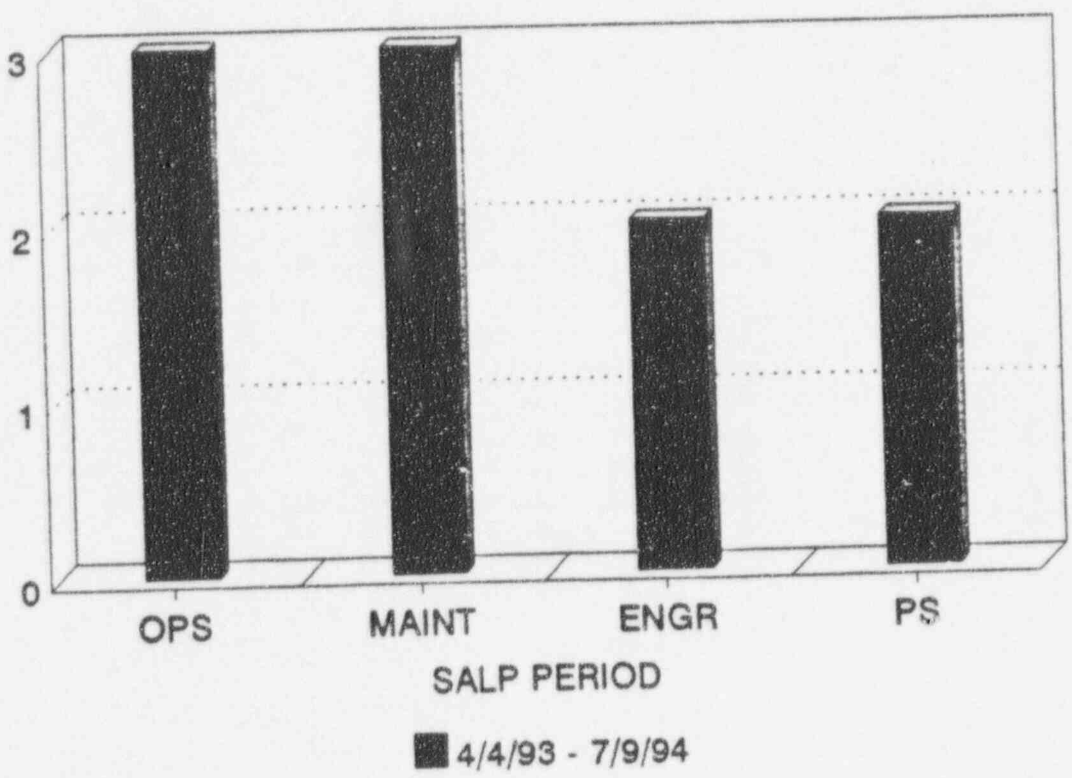
PENDING This action is based on the potential loss of ECCS during the recirculation phase of a LOCA due to a thermal expansion pressure locking of the containment sump suction valves. An enforcement conference on this matter was held on April 18, 1995.

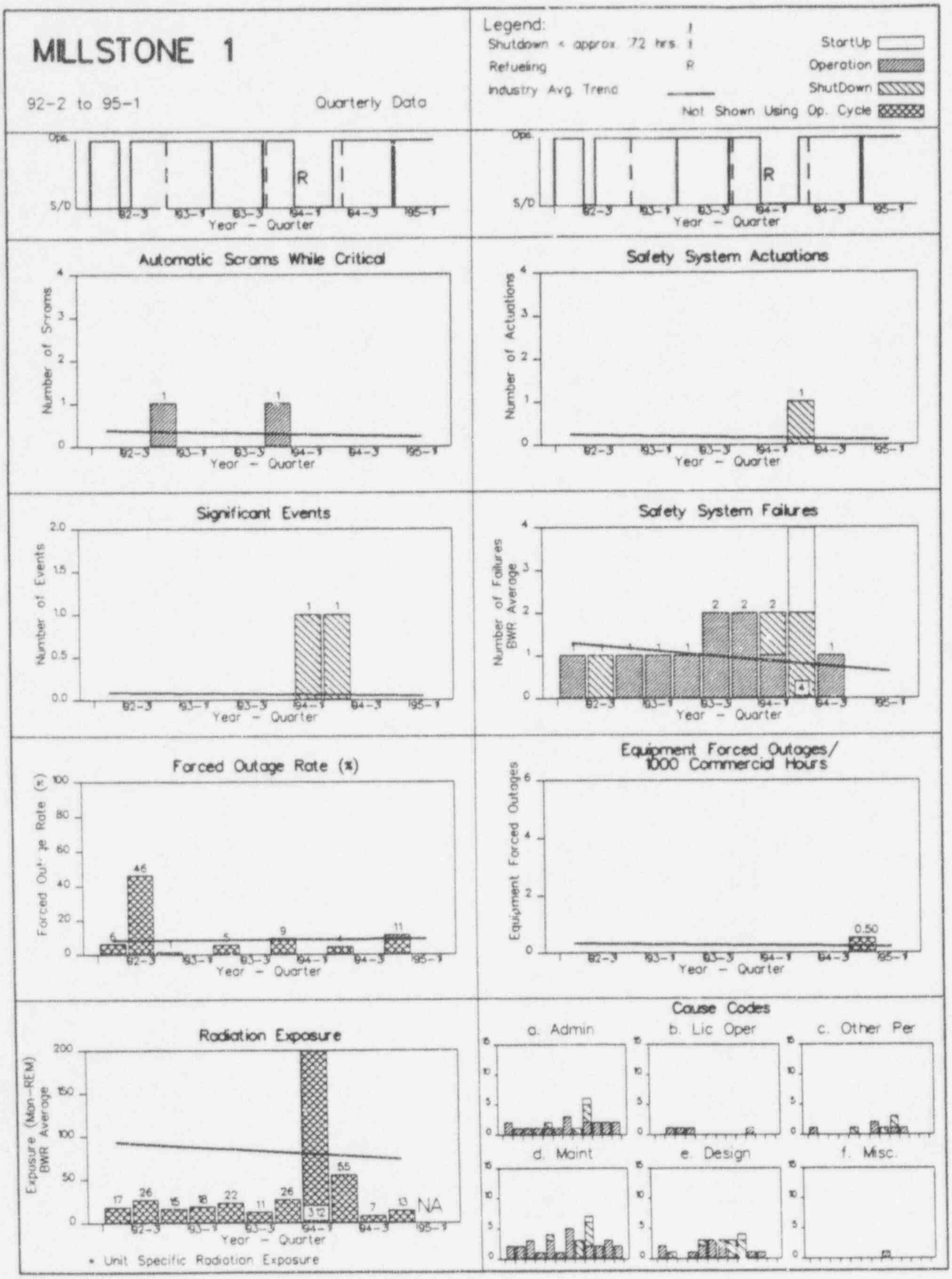
MILLSTONE 2

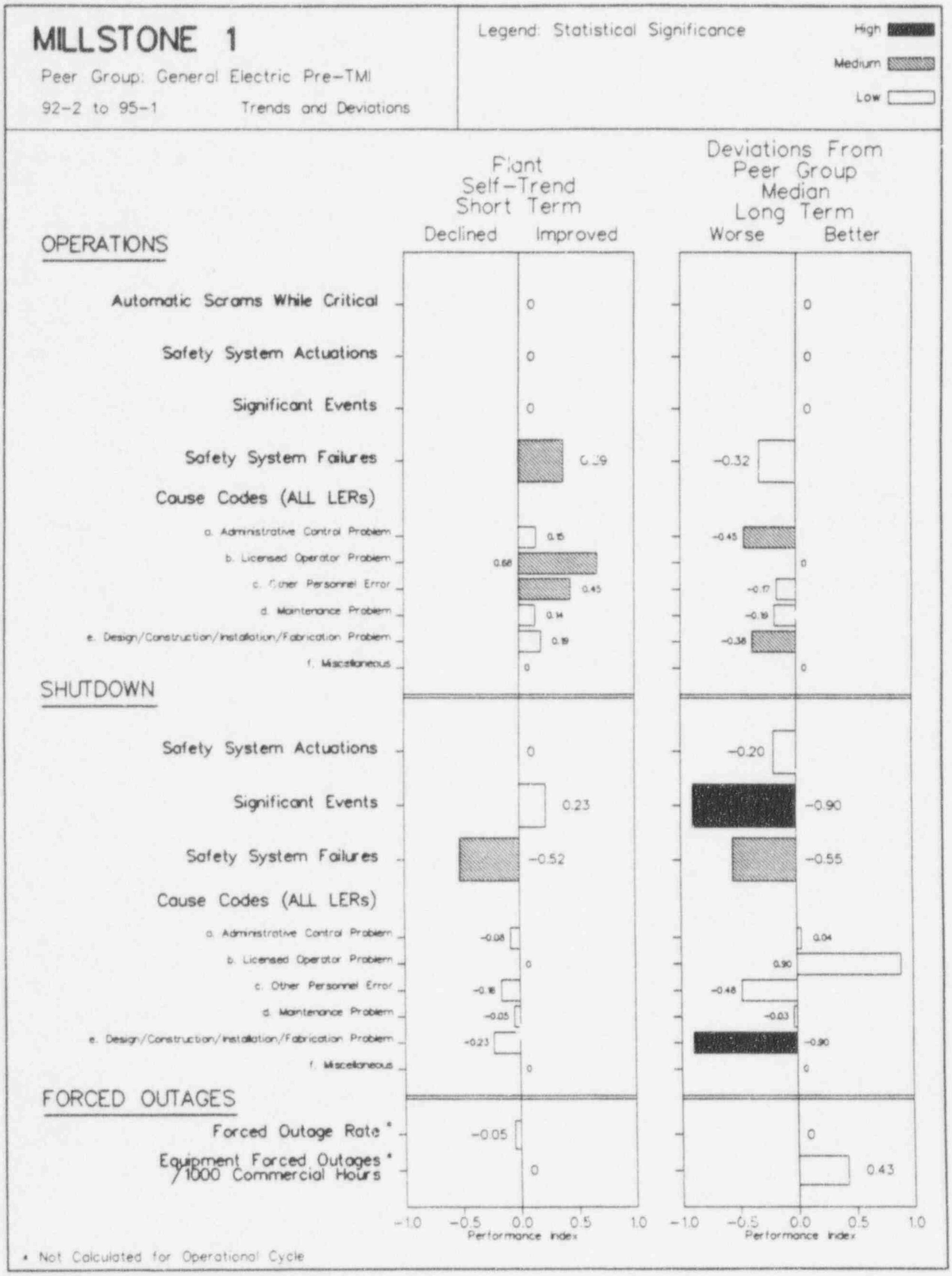
MOST RECENT SALP RATINGS

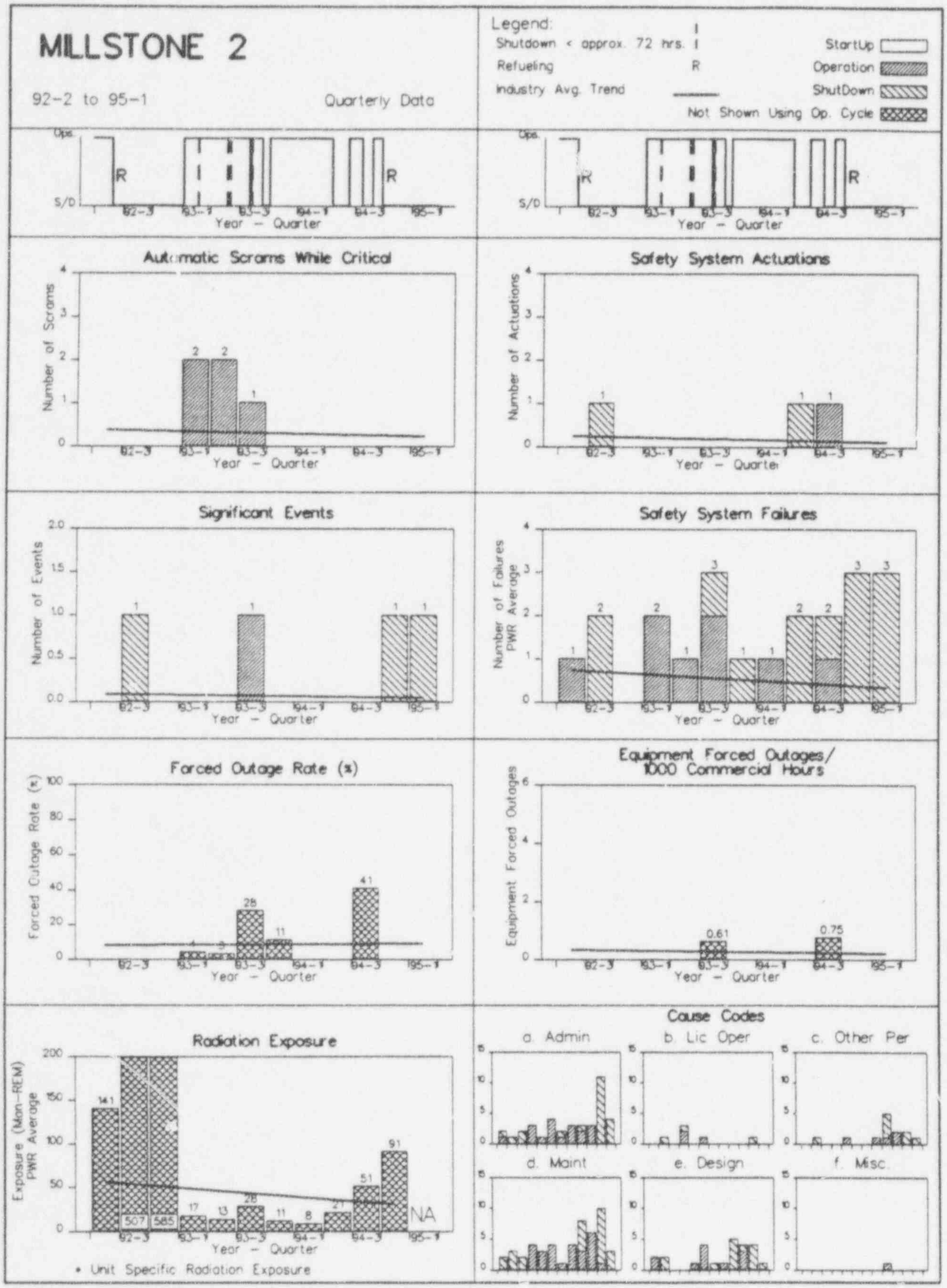


REVISED SALP PROGRAM RATINGS

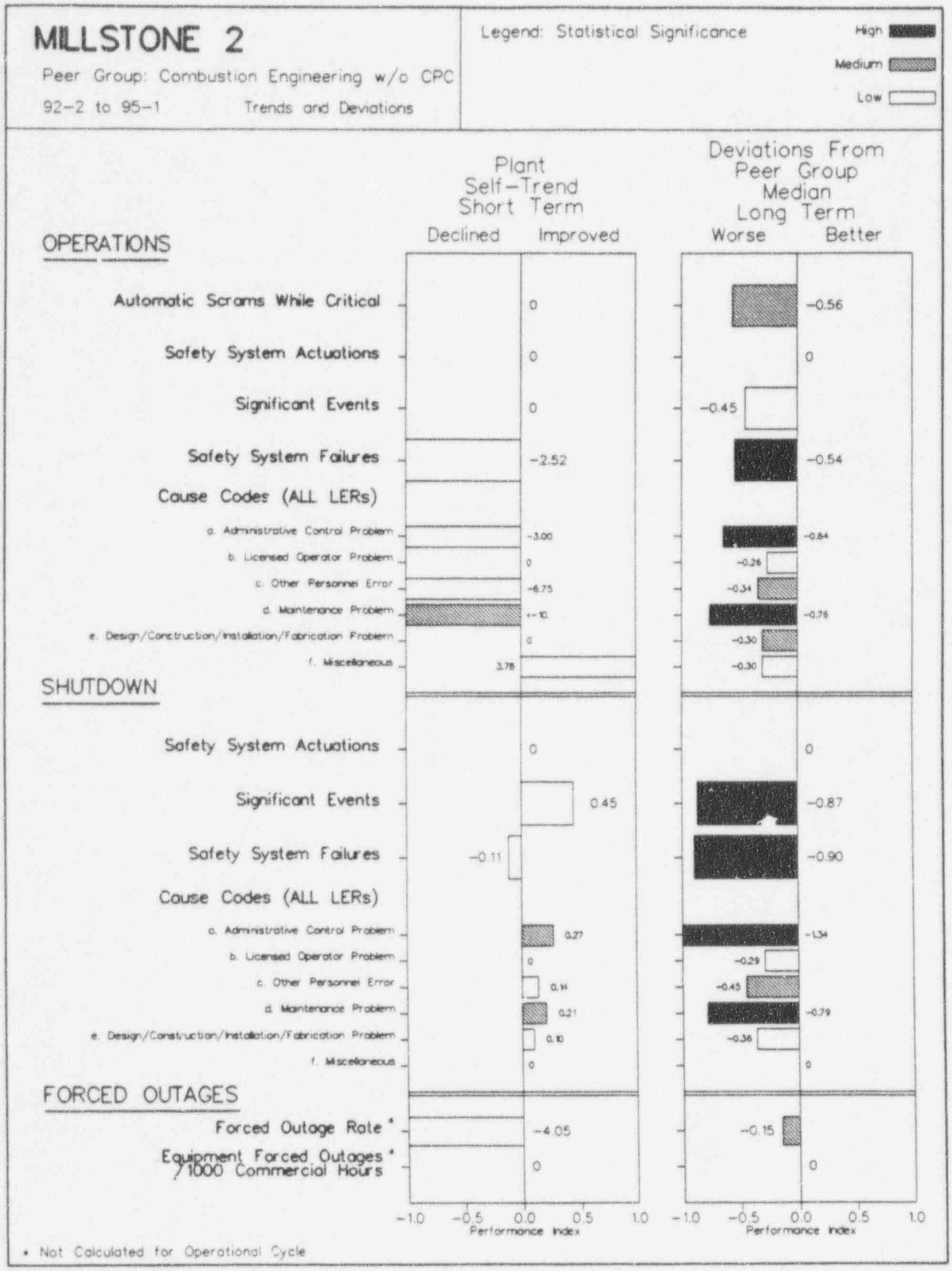


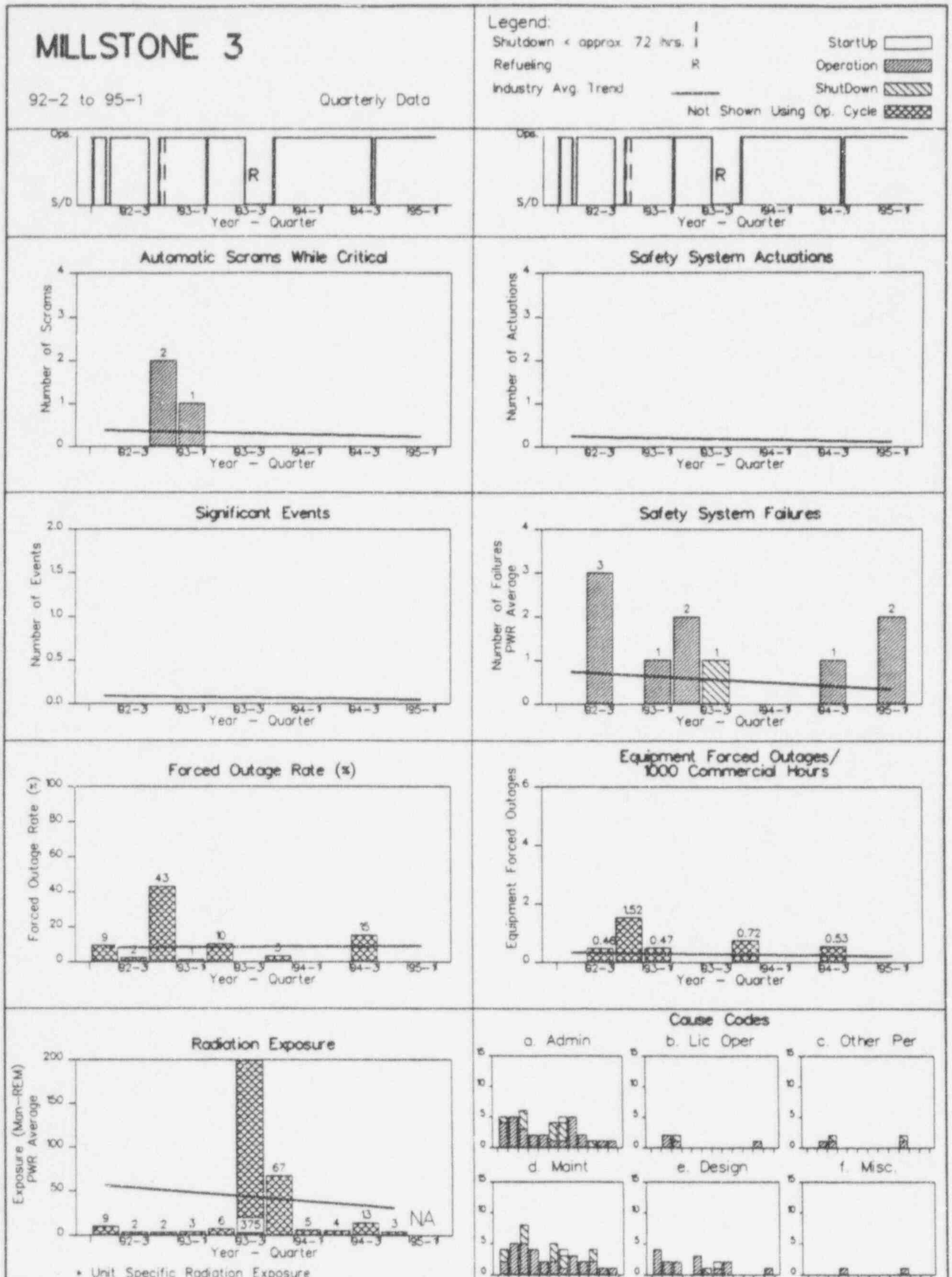

MILLSTONE 1
92-2 to 95-1
Quarterly Data
Legend:
Shutdown < approx. 72 hrs.
Refueling
R
Industry Avg. Trend
StartUp
Operation
ShutDown
Not Shown Using Op. Cycle
Year - Quarter
Automatic Scrams While Critical
Number of Scrams
Safety System Actuations
Number of Actuations
Significant Events
Number of Events
Safety System Failures
Number of Failures
BWR Average
Forced Outage Rate (%)
Equipment Forced Outages/
1000 Commercial Hours
Equipment Forced Outages
Radiation Exposure
Exposure (Man-REM)
NA
* Unit Specific Radiation Exposure
Cause Codes
a. Admin
b. Lic Oper
c. Other Per
d. Maint
e. Design
f. Misc.

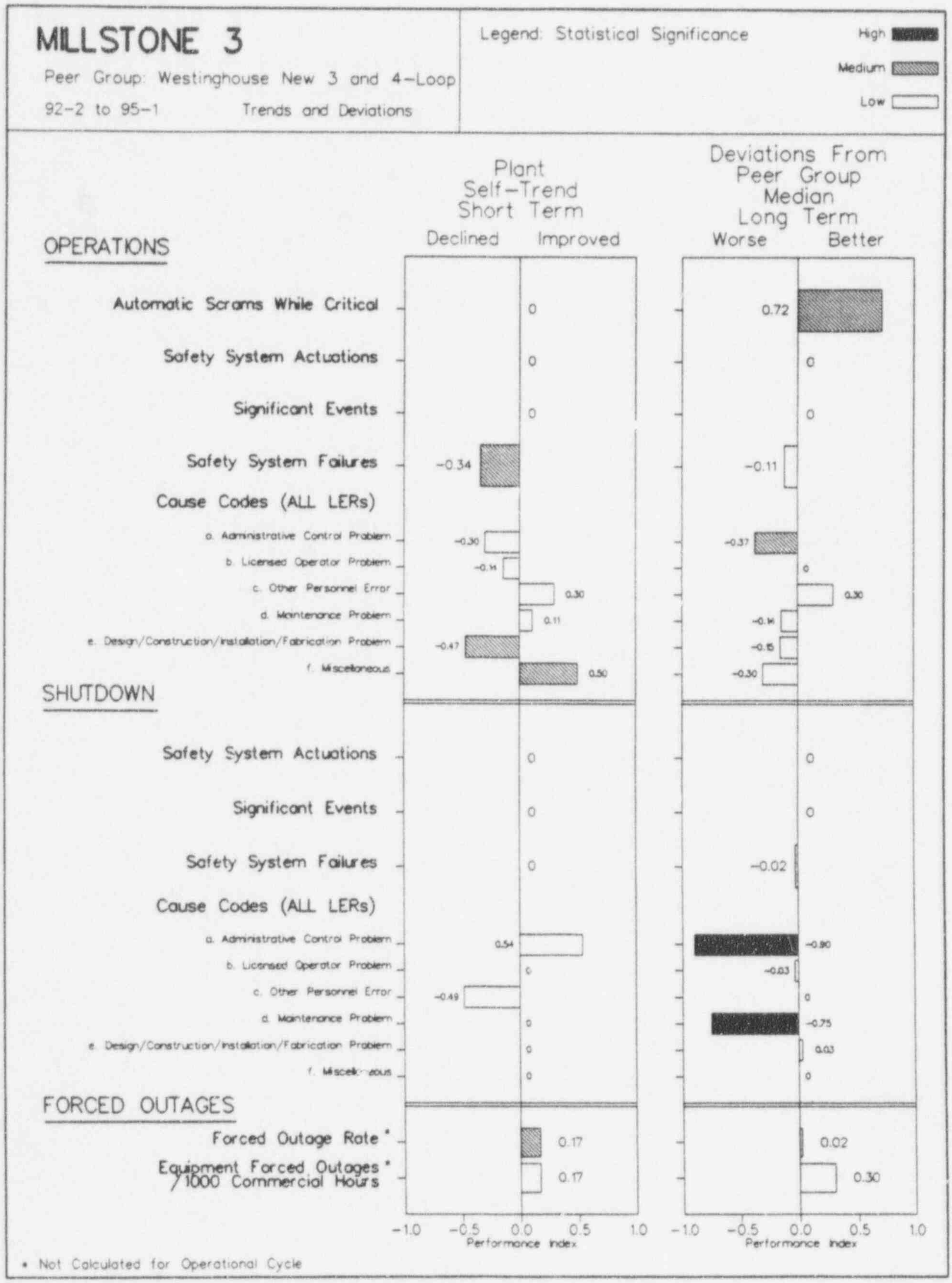

MILLSTONE 1
Peer Group: General Electric Pre-TMI
92-2 to 95-1 Trends and Deviations
Legend: Statistical Significance
High
Medium
Low
Plant Self-Trend Short Term
Declined Improved
Deviations From Peer Group Median Long Term
Worse Better
OPERATIONS
Automatic Scrams While Critical
Safety System Actuations
Significant Events
Safety System Failures
Cause Codes (ALL LERs)
a. Administrative Control Problem
b. Licensed Operator Problem
c. Other Personnel Error
d. Maintenance Problem
e. Design/Construction/Installation/Fabrication Problem
f. Miscellaneous
SHUTDOWN
Safety System Actuations
Significant Events
Safety System Failures
Cause Codes (ALL LERs)
a. Administrative Control Problem
b. Licensed Operator Problem
c. Other Personnel Error
d. Maintenance Problem
e. Design/Construction/Installation/Fabrication Problem
f. Miscellaneous
FORCED OUTAGES
Forced Outage Rate *
Equipment Forced Outages * /1000 Commercial Hours
-0.32
-0.20
0.23
-0.90
-0.52
-0.55
-0.05
0.43
Performance Index
* Not Calculated for Operational Cycle



MILLSTONE 2
92-2 to 95-1
Quarterly Data
Legend:
Shutdown < approx. 72 hrs.
Refueling R
Industry Avg. Trend
StartUp
Operation
ShutDown
Not Shown Using Op. Cycle
Year - Quarter
Automatic Scrams While Critical
Number of Scrams
Safety System Actuations
Number of Actuations
Significant Events
Number of Events
Safety System Failures
Number of Failures PWR Average
Forced Outage Rate (%)
Forced Outage Rate (%)
Equipment Forced Outages/ 1000 Commercial Hours
Equipment Forced Outages
Radiation Exposure
Exposure (Man-REM) PWR Average
NA
* Unit Specific Radiation Exposure
Cause Codes
a. Admin
b. Lic Oper
c. Other Per
d. Maint
e. Design
f. Misc.


MILLSTONE 2
Peer Group: Combustion Engineering w/o CPC
92-2 to 95-1
Trends and Deviations
Legend: Statistical Significance
High
Medium
Low
Plant Self-Trend Short Term
Declined
Improved
Deviations From Peer Group Median Long Term
Worse
Better
OPERATIONS
Automatic Scrams While Critical
0
-0.56
Safety System Actuations
0
0
Significant Events
0
-0.45
Safety System Failures
-2.52
-0.54
Cause Codes (ALL LERs)
a. Administrative Control Problem
-3.00
-0.84
b. Licensed Operator Problem
0
-0.26
c. Other Personnel Error
-6.75
-0.34
d. Maintenance Problem
<-10.
-0.76
e. Design/Construction/Installation/Fabrication Problem
0
-0.30
f. Miscellaneous
3.78
-0.30
SHUTDOWN
Safety System Actuations
0
0
Significant Events
0.45
-0.87
Safety System Failures
-0.11
-0.90
Cause Codes (ALL LERs)
a. Administrative Control Problem
0.27
-1.34
b. Licensed Operator Problem
0
-0.29
c. Other Personnel Error
0.14
-0.45
d. Maintenance Problem
0.21
-0.79
e. Design/Construction/Installation/Fabrication Problem
0.10
-0.36
f. Miscellaneous
0
0
FORCED OUTAGES
Forced Outage Rate *
-4.05
-0.15
Equipment Forced Outages * /1000 Commercial Hours
0
0
-1.0 -0.5 0.0 0.5 1.0
Performance Index
* Not Calculated for Operational Cycle


MILLSTONE 3
92-2 to 95-1
Quarterly Data
Legend:
Shutdown < approx. 72 hrs.
Refueling R
Industry Avg. Trend
StartUp
Operation
ShutDown
Not Shown Using Op. Cycle
Ops.
S/D
R
Year - Quarter
92-3
93-1
93-3
94-1
94-3
95-1
Automatic Scrams While Critical
Number of Scrams
2
1
Safety System Actuations
Number of Actuations
Significant Events
Number of Events
Safety System Failures
Number of Failures
PWR Average
3
1
2
1
1
2
Forced Outage Rate (%)
Forced Outage Rate (%)
9
2
43
10
3
15
Equipment Forced Outages/
1000 Commercial Hours
Equipment Forced Outages
0.46
1.52
0.47
0.72
0.53
Radiation Exposure
Exposure (Man-REM)
PWR Average
9
2
2
3
6
375
67
5
4
13
3
NA
* Unit Specific Radiation Exposure
Cause Codes
a. Admin
b. Lic Oper
c. Other Per
d. Maint
e. Design
f. Misc.


MILLSTONE 3
Peer Group: Westinghouse New 3 and 4-Loop
92-2 to 95-1
Trends and Deviations
Legend: Statistical Significance
High
Medium
Low
Plant Self-Trend Short Term
Declined
Improved
Deviations From Peer Group Median Long Term
Worse
Better
OPERATIONS
Automatic Scrams While Critical
Safety System Actuations
Significant Events
Safety System Failures
Cause Codes (ALL LERs)
a. Administrative Control Problem
b. Licensed Operator Problem
c. Other Personnel Error
d. Maintenance Problem
e. Design/Construction/Installation/Fabrication Problem
f. Miscellaneous
SHUTDOWN
Safety System Actuations
Significant Events
Safety System Failures
Cause Codes (ALL LERs)
a. Administrative Control Problem
b. Licensed Operator Problem
c. Other Personnel Error
d. Maintenance Problem
e. Design/Construction/Installation/Fabrication Problem
FORCED OUTAGES
Forced Outage Rate *
Equipment Forced Outages * /1000 Commercial Hours
0
-0.34
-0.30
-0.14
0.30
0.11
-0.47
0.50
0.54
-0.49
0.17
0.72
-0.11
-0.37
-0.15
-0.02
-0.80
-0.03
-0.75
0.03
0.02
-1.0
-0.5
0.0
0.5
1.0
Performance Index
* Not Calculated for Operational Cycle

NORTHEAST NUCLEAR ENERGY COMPANY
MILLSTONE 1, 2, & 3