U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-219/90-12 License No. DPR-16

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

Facility Name: Oyster Creek Nuclear Generating Station

Inspection Conducted: July 12, 1990 - August 22, 1990

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Reactor Projects Section 4B

Inspection Summary:
Inspection Report No. 50-219/90-12 for July 12, 1990 - August 22, 1990

Areas Inspected: The inspection consisted of 250 hours of direct inspection. The areas inspected included observation and review of plant operational events (section 1.0); review of radiological events (section 2.0); routine observations of maintenance activities and surveillance tests (section 3.0); review of emergency diesel generator surveillance test failures (section 4.0); and review of licensee critique and corrective actions for several plant events and incidences (section 6.0).

Results: An executive summary is enclosed with the report. An unresolved item related to possible operation above the licensed reactor power due to errors in feedwater flow calibration is opened in this report.

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*The NRC inspection manual inspection procedure (IP) or temporary instruction (TI) that was used as inspection guidance is listed for each applicable report section.

ATTACHMENTS

Attachment I: Sequence of Events for August 6 Unusual Event
Attachment II: Presentation for Commissioner Curtiss' Visit

U. S. NUCLEAR REGULATORY COMMISSION
REGION I
EXECUTIVE SUMMARY

Report No. 50-219/90-12

Operations

Overall the plant was operated in a safe manner. Licensee action to declare an Unusual Event on August 6, 1990, when unidentified leakage exceeded 5 gpm was appropriate. Licensee review identified the source of water to be a containment spray valve which was not fully closed during the performance of a containment spray system surveillance test. The operators vented the drywell at 2:42 p.m. due to increasing pressure following the water addition. The inspectors concluded that the operators' actions were appropriate, in accordance with station procedures, and steps taken by the licensee to ensure the safety of the plant were adequate.

Radiological Controls

At about 3:33 p.m. on August 6, 1990, the State of new Jersey observed elevated readings on its radiation monitors located northeast of the Oyster Creek site at a distance of 1.3 miles. At that time, Oyster Creek was in an Unusual Event due to increased unidentified leak rate and the drywell was vented due to elevated pressure. NRC inspectors reviewed the time of drywell venting, the amount of radioactivity in the drywell atmosphere, meteorological and plant conditions at the time of the drywell venting, normal plant stack releases, and the response of in-plant radiation monitors. The inspectors concluded the venting of the drywell had a negligible radiological impact offsite and onsite activities did not cause the elevated monitor readings.

Maintenance/Surveillance

The inspectors reviewed the station procedure for control rod scram time testing. The procedure was found appropriate. During scram time testing, the charging water header is isolated so the effects of the Control Rod Hydraulic pump are removed from the scram.

Engineering and Technical Support

During a surveillance test on July 9, 1990, the licensee identified a degraded battery cell in No. 2 emergency diesel generator. In April 1990, a degraded battery cell was found in No. 1 emergency diesel generator. Licensee evaluation concluded these degraded cells were early signs of the end of battery life. In both cases, the degraded cell did not affect the capability of the engine to start in emergency conditions. As a result, the licensee decided to replace the batteries prior to the upcoming (13R) refueling outage, if possible.

On August 15, 1990, the No. 2 emergency diesel generator had load swings during a surveillance test. The licensee identified the cause to be a loose sub-governor case inside the governor actuator. The inspecto^ concluded the loosening of this subgovernor housing is a long term effect and was adequately captured by the licensee's surveillance program. Licensee plans to inspect this component during refueling outages are appropriate.

Safety Assessment/Quality Verification

The tensiometer used in installation of 1B2 cable du~ing 12U-K outage was found out of calibration. The licensee concluded that cable tension exceeded the manufacturer's specified maximum pull tension during installation. The licensee reviewed the results of field tests done on the installed cable and laboratory tests done on a sample of cable that was pulled through the conduit. Based on these test results the installed caule was evaluated as acceptable. The licensee is developing a periodic test program for the installed 4.16 kv cables to be implemented during the 13R ref.,ling outage.

DETAILS

1.0 Plant Operational Review

1.1 Chronology of Operational Events

Inspectors reviewed details associated with key operational events that occurred during the report period. A summary of these inspection activities follows.

-- 7/12/90 The inspection period started with the plant in a seven-day technical specification limiting condition for operation (LCO) that started on 7/9/90 with the No. 2 emergency diesel generator out of service.

The reactor was at 66.9 percent of licensed power level of 1930 MWth. The reduction in power was necessary to clean grass in the plant intake that migrated through the screen and affected performance of the No. 4 circulating water pump. Reactor power was increased to full power following grass removal.

-- 7/13/90 The No. 2 emergency diesel generator was declared operable after replacement of a degraded battery cell and necessary adjustments. Section 4.1 discusses these corrective actions.

-- 7/17/90 The operators inserted control rod 18-07 to the full in "00" position from full out "48" position. Due to a leaking charging valve V-111, the hydraulic control unit (HCU) could not be maintained charged and was isolated. The accumulator low pressure alarm was bypassed and the control rod declared inoperable. After V-111 valve was replaced, the control rod was declared operable and withdrawn to its programmed position of 48.

-- 7/25/90 The licensee commenced a reactor shutdown as required by the plant technical specification after "D" main steam line radiation monitor was found out of calibration during a surveillance test. The licensee declared the monitor inoperable. The reactor shutdown was required per technical specification when the trip module was reset for troubleshooting and repair. The radiation monitor was recalibrated, declared operable and reactor shutdown terminated the same day after approximately three hours.

-- 7/31/90 Reactor power was reduced to approximately 48 percent with two out of four circulating water pumps taken out of service. Due to migration of grass into the intake, the south intake suction pressure was low. The emergency service water (ESW) pumps 52 C and D take suction from the south side of the plant intake, and as a conservative measure the licensee declared the ESW system No. 2 inoperable. After cleanup of grass and necessary repairs to the intake screens, the circulating water pumps were placed back in service, power increase commenced and ESW system No. 2 was declared operable.

-- 8/2/90 The service water radiation monitor was declared operable after a long period of inoperability.

-- 8/6/90 During a containment spray system surveillance test, the licensee inadvertently introduced about 300 gallons of water into the drywell due to the discharge valve for system No. 2 not fully closing. A seven-day technical specification action statement was started due to an inoperable drywell discharge valve on system No 2 while the licensee continued troubleshooting the reason for the valve not fully closing. A description of the event and the licensee's review and corrective action are described in section 1.2.

-- 8/8/90 Containment spray system No. 2 was declared operable after surveillance testing.

-- 8/13/90 Emergency diesel generator No. 2 was declared inoperable and a seven-day technical specification LCO was initiated due to load swings observed during a load test. Details of the event, the licensee's review and corrective actions are described in section 4.2.

The augmented offgas system (AOG) tripped during a lightning strike which also damaged several reflash units in a panel in the control room. The reflash units were isolated and later replaced. The AOG system was returned to service. Similar lightning damages have occurred in the past (see inspection report 50-219/90-09). The licensee is currently reviewing the electrical systems for necessary surge protection.

-- 8/20/90 Due to a leaking valve (V-III), HCU 10-43 was found unable to maintain pressure. The control rod was declared inoperable. An engineering evaluation demonstrated compliance with technical specification required shutdown margin. The licensee replaced the valve and declared the control rod operable after approximately six hours.

-- 8/22/90 The licensee identified a possible unmonitored release path on the turbine building northwest roof. An open drain line from the the reheater protection system admitted steam to the building floor drain system and ultimately to the turbine building roof via temporary piping. The release was terminated by blocking the temporary piping. Sample results indicated concentrations to be well below the regulatory limits. Inspection Report 50-219/90-13 reviewed the environmental consequences of this release. At the end of the inspection period, resident inspectors were reviewing the event and the licensee's corrective actions.

-- At various times during this period, the licensee reduced reactor power level to perform condenser backwashing, to maintain condenser vacuum or condenser discharge temperature limits when intake water temperature was high, and due to migration of grass into the intake bay which reduced circulating water pump suction pressure.

1.2 Unusual Event with High Unidentified Drywell Leakage

Event Description

On August 6, 1990, the licensee performed surveillance test 607.3.002, Rev. 34, Containment Spray Automatic Actuation. After testing drywell pressure switches IP15B and D, the actuation logic was tested by inserting a start signal (about 2:37 p.m.) and verifying operation of containment spray pump 51C (pump started about 45 seconds later). About one minute after pump start, drywell unidentified leakage indicated high. Then, drywell cooler outlet temperature alarms (115 degrees F) were received. The containment spray pump was secured (2:40:15 p.m.). At 2:40:45 p.m., a high drywell pressure alarm was received (alarm setpoint of 1.4 psig).

In response to the high drywell temperature alarms, Control Room operators verified that all Electromatic Relief Valves (EMRV) and safety valves indicated closed, verified that downcomer temperatures were normal, observed that drywell humidity had increased from about 30 to 50 percent, and started the remaining drywell fan.

In response to the increasing drywell pressure, operators vented the torus and drywell to the main stack using a two inch bypass line. It was verified that stack activities did not increase during this evolution.

An Unusual Event was declared based on the indicated high drywell unidentified leakage at 3:15 p.m. and a plant shutdown was started.

A detailed sequence of events is included as Attachment I.

Adequacy of Plant Procedures and Operator Actions

NRC inspectors reviewed the relevant plant procedures used during the event to determine their adequacy. Inspectors also interviewed certain control room operators present during the event to determine the appropriateness of their actions. Inspectors concluded that operator actions were appropriate and in accordance with the existing plant procedures. However, one procedure needs enhancement to proceduralize the need to monitor stack radioactivity indications during routine venting of the drywell and torus.

About 300 gallons of torus water were introduced into the drywell because of incomplete closure of containment spray valve V-21-5. This water flashed into vapor. This vapor caused the increase in drywell cooler outlet temperatures and caused the increase in drywell pressure from its normal value of about 1.1 psig to a maximum value of 1.57 psig.

Alarm Response Procedure C-8-h, DW TEMP HI (setpoint 115 degrees F), revision 27, directed the operator to refer to Drywell Cooling System Diagnostic and Restoration Actions Procedure 2000-OPS-3024.09 for corrective actions. It also directed the operator to check for a drywell leak by monitoring drywell humidity, torus water level, reactor water level, condensate storage tank level, relief/safety valve discharge temperatures, unidentified leak rate, and drywell bulk temperature.

To return pressure to its normal operating range, Alarm Response Procedure C-3-f, DW Pressure Hi/Lo (setpoint 1.4/1.0 psig), Revision 19, directed the operator to vent the drywell and torus per Station Procedure 312, "Reactor Containment Integrity and Atmosphere Control, Revision 45."

The operators monitored the radiation indications in the reactor building ventilation exhaust and the stack radioactive gas effluent monitor to verify no increase of radioactivity during the evolution. Operators performed this verification even though there was no caution or procedural step to this effect. The licensee committed to revise station procedure 312 to include this requirement. This is to be completed in September 1990.

Technical specification table 3.15.2, item 2.a., Action 124, allows drywell purge only when the radioactive noble gas monitor is operable. This requirement ensures measurement of the large volume of radioactive gases discharged. The operability of the monitor is not required for the two inch drywell vent path used during this event. This vent path is routinely used to control drywell pressure. Although unfiltered, the drywell atmosphere is diluted by a factor of about 500 by the turbine and reactor building effluents before leaving the stack. Technical specification bases page 3.15-3 indicates that because the release rate associated with normal drywell venting is small compared to the drywell purge, and the effluent is monitored as usual, the requirement in Table 3.15.2 Action 124 that is applied during drywell purging is not imposed during drywell venting. Since the effluent from the drywell venting evolution is of relatively small volume, is diluted, and uses an elevated release point, the existing procedure (312) is adequate to protect the health and safety of the public.

During this event, the drywell was vented through an unfiltered path as directed by procedure. Although acceptable, the licensee is evaluating the need for additional guidance to direct the use of the Standby Gas Treatment System.

NRC inspectors concluded operator response to this event was acceptable and in accordance with station procedures. Procedures are adequate, but can be enhanced as described above.

Root Cause of the Event

Evaluation by the licensee's post transient review group (PTRG-90-135A) concluded the root cause was a design configuration deficiency. Procedure 607.3.002 directed the operators to deenergize valve V-21-5 after it indicated closed. This valve indicates closed when it is about 80 percent closed, and requires about 15 more seconds to reach 100 percent closed. In this event, operators deenergized the valve about five seconds after receiving the closed indication, but before it reached full closed. This cause was confirmed by a review of the applicable computer records. This permitted water to be introduced into the drywell.

The licensee had generated a modification to use a different limit switch rotor for valve position indication. This allows the indicating light to be adjusted independently of the opening torque switch bypass limit switch. Already implemented on many valves, this modification has been scheduled for the next refueling outage for containment spray valves. The licensee plans to implement this modification at the earliest opportunity on Containment Spray system valves

Even though control room operators have been trained on this modification, and told that valves may require more travel time after indicating closed, this was not recognized during this event. As an interim corrective action, a memorandum was issued to all operations and maintenance personnel to remind them that motor operated valves will require additional time to reach full closure after the remote closed indication is received.

Equipment Review

To review the impact of this water on drywell equipment, the PTRG reviewed the consequences of a similar event that occurred in 1982. NRC review of this event is documented in Inspection Report 50-219/82-29.

On December 21, 1982, an operator mistakenly started a containment spray pump aligned to the drywell and sprayed 2000 to 3000 gallons of water into the drywell in approximately 60 seconds. At that time, the licensee, in conjunction with the reactor manufacturer, conducted a full investigation of the event to evaluate its impact on electrical and mechanical equipment inside the drywell and to determine the tests which may be necessary to ensure the safety of the plant.

That investigation concluded that there was no degradation to the integrity of the piping, valves, or other mechanical components required to perform a safety function as a result of the chlorides or chromates contained in the spray water or as a result of the thermal stresses.

With respect to electrical equipment in 1982, the licensee prepared a list of all components within the drywell, including junction boxes and wiring devices and, on the basis of required safety functions and direct observation of equipment status, determined that it would be necessary to perform the following surveillance/operability tests: (1) full closure of main steam isolation valves and operability of limit switches; (2) operability of reactor water sample solenoid operated isolation valves and limit switches; (3) operability of isolation condenser motor operated valves and limit switches; (4) resistance measurements of solenoid coil and insulation for the Electromatic relief valve solenoid operators and limit switches; and (5) surveillance of the acoustic valve monitoring system. An evaluation of the results of these tests concluded that the inadvertent containment spray actuation had produced no detrimental consequences on the safety related equipment affected.

Following the current event, the licensee reviewed the results of the 1982 analysis and concluded that a similar approach to the concern would be sufficient to establish the safe status of the plant. For this purpose, the licensee initiated some of the above tests while they evaluated all modifications that were performed since 1982 and that affected the equipment in the drywell. This review concluded that surveillance tests of the hydrogen/oxygen analyzer were warranted. With regard to the 1982 tests, the licensee concluded that resistance measurements of the solenoid coil and insulation for the electromatic relief valve solenoid operators and switches were not necessary. The basis for the decision was: (1) the positive results obtained from the 1982 tests; (2) the lesser magnitude of the current event; (3) the protection afforded by the steel enclosure with only bottom opening; (4) replacement of the old solenoids with new ones encapsulated and qualified to current standards; (5) use of qualified connectors; and (6) smaller amount of steam generated from the spray.

The inspector evaluated the environmental conditions recorded following the water introduction, the results of the licensee's analysis, and the results of the surveillance/operability tests performed and concluded that the steps taken by the licensee to ensure the safety of the plant were adequate.

Inspectors reviewed the containment response to the water introduction. Drywell pressure increased by about .3 psi while torus pressure did not change. This demonstrated integrity of the torus - drywell vacuum breakers.

Inspectors questioned the possible impact of wetting of pipe insulation. The licensee stated that in refueling outage 10R, asbestos insulation on stainless steel piping was replaced with Nukon insulation. This type of insulation is not susceptible to chloride leaching upon wetting. Inspectors had no other questions about the wetting of pipe insulation.

Conclusions

Overall, licensee response to this event was acceptable from a safety perspective. Control Room operator response was appropriate and in accordance with station procedures. Station procedures were adequate. Site Emergency Operating Procedures were not required for this event. The possible detrimental effects of water on safety related equipment were adequately evaluated. No plant equipment malfunctions have been observed. Corrective actions were adequate.

1.3 Control Room Tours

The inspectors conducted routine tours of the control room. The inspectors reviewed:

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

-- Technical Specification Log;

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

-- Reactor Building and Turbine Building Tour Sheets;

-- Equipment Control Logs;

-- Standing Orders; and,

-- Operational Memos and Directives.

No significant observations were identified.

1.4 Facility Tours

The inspectors conducted routine plant tours to assess equipment conditions, personnel safety hazards, procedural adherence and compliance with regulatory requirements. The following areas were inspected:

-- Turbine Building

-- Vital Switchgear Rooms

-- Cable Spreading Room

-- Diesel Generator Building

-- Reactor Building

-- New Radwaste Building

-- Old Radwaste Building

-- Plant Intake Area

The following additional items were observed or verified:

a. Fire Protection:

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

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

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

-- Appropriate fire watches or fire patrols were stationed when fire protection/detection equipment or fire barriers including doors were out of service.

b. Vital Instrumentation:

-- Selected instruments appeared functional and demonstrated parameters within Technical Specification Limiting Conditions for Operation.

c. Housekeeping:

-- Plant housekeeping and cleanliness were in accordance with licensee programs.

Minor housekeeping deficiencies which were identified were promptly corrected by the licensee. No other unacceptable conditions were identified.

2.0 Radiological Controls

Offsite Impact of Venting the Drywell on August 6, 1990:

In the afternoon of August 6, 1990, the State of New Jersey observed elevated readings on its Reuter-Stokes pressurized ion chamber located northeast of the plant at a distance of about 1.3 miles. The elevated readings that occurred were as follows:

Time	μR/hr (1 microRem = .000001 Rem)
1533	background
1534	128
1535	114
1536	10
1537	background

Time	μR/hr (1 microRem = .000001 Rem)
1611	background
1612	11
1613-1633	about 45
1634	background

Review of licensee's data from their onsite meteorological tower revealed that from about 2:42 p.m. on August 6 through about 4:45 p.m., the direction of the ground-level wind was toward the North. During this same time, the wind direction at the height of the stack (380' elevation) was toward the North-Northeast. Both the ground-level and the elevated wind direction data exhibited considerable variation. The ground-level wind velocity was about 5 mph and the elevated wind velocity was about 8 mph.

Normal noble gas releases from the plant stack were present on the afternoon of August 6, 1990, and were about 20 uCi/sec, with the majority of this activity coming from the Augmented Offgas System. The path for the drywell vent was also via the plant stack. The drywell atmosphere was sampled on August 8, 1990. The predominant radionuclides were xenon-133 and xenon-135. The concentrations of these radionuclides in the drywell atmosphere were 4.6E-6 uCi/ml and 4.2E-6 uCi/ml, respectively. The volume of containment gas that was released over the 17 minute venting period was estimated at 3870 cubic feet based on the observed changes in drywell pressure. This volume was averaged over the 17 minute venting period. The estimated noble gas release rate that occurred as a result of the drywell vent was about 1 uCi/sec.

Review of the licensee's monitors located on the stack and at the Reactor Building vent duct indicated no measurable increase in noble gas activity, iodines, or particulates as a result of the drywell purge. A 1 uCi/sec increase in noble gas release rate would be masked by the fluctuations of the normal 20 uCi/sec noble gas release rate measurement.

Conclusion:

The direction of the wind, while not directly toward the affected offsite monitor, was in the general direction and, in view of the broad variation in the data, wind direction is considered to support a correlation of a plant event with the monitor response. However, regarding the venting of the drywell, two pieces of information strongly do not support a correlation between the drywell purge and offsite monitor response:

-- First, as indicated above, the activity release rate from the drywell was very small relative to normal plant release rates that were occurring at the time (1 uCi/sec from the drywell purge versus 20 uCi/sec from the plant).

-- Second, the elevated and ground-level wind speed are inconsistent with the time of the offsite monitor indications relative to when the drywell vent occurred. The elevated wind speed was about 8 mph. The plume from the purge would have been expected to reach the offsite monitor in about 10 minutes or about 2:52 p.m. and stay there for about 20 minutes. The offsite monitor indications occurred about 3:32 p.m. for a duration of 3 minutes, and then around 4:12 p.m. for about 20 minutes.

The inspectors concluded that the response of the offsite monitor was unrelated to the venting of the drywell that occurred on August 6, 1990.

3.0 Maintenance/Surveillance

3.1 Control Rod Scram Time Testing Methodologies

Inspectors reviewed Station Procedure 617.4.003, Rev. 12, "Control Rod Scram Insertion Time Test and Valve IST Test," to determine if the effects of the control rod drive hydraulic pumps are considered in control rod scram time testing. The inspector observed that procedure 617.4.003 requires closing of the charging water valve V-106 for the selected control rod prior to scram time testing. In addition, prerequisite 3.5 requires reactor pressure to be greater than 800 psi. When the control rod is scrammed with the charging water header isolated, the accumulator will discharge, allowing reactor pressure to complete the scramming of the rod, thus verifying scram time in the worse case condition and correct operation of a ball check valve inside the control rod drive mechanism. The inspector had no further observations.

3.2 Containment Spray Automatic Actuation Test

On August 6, 1990, inspectors observed performance of Surveillance Test Procedure 607.3.002, Rev. 34, "Containment Spray Automatic Actuation." Inspectors observed instrument and control technicians testing and adjusting drywell pressure switches IP15B and IP15D. Inspectors verified that the technicians were performing the test and adjustments in accordance with procedural instructions and that the test results were properly documented. Inspectors also observed operation of containment spray pump 51C. A minor packing leak was observed on containment spray pump suction valve V-21-1. This leak was reported to the control room. The inspector had no further observations.

3.3 Core Spray System No. 2 Instrument Channel Calibration and Test

On July 19, 1990, the inspector observed performance of Surveillance Test Procedure 610.3.205, Rev. 16, "Core Spray System 1 Instrument Channel Calibration and Test," for high drywell pressure sensors RV46C and D. The inspector verified that the technicians had the appropriate approvals and were following the procedure and that the test equipment and gauges were appropriate and calibrated. No unacceptable conditions were identified.

3.4 Analysis of Discharge and Intake Canal Effluents

The inspector reviewed the results for the licensee's effluent analysis of the discharge and intake canal for pH and residual chlorine. Station Procedure 828.3, Rev. 14, "Secondary System Analysis: Plant Effluents/NJ PDES," requires periodic sampling and analysis of the 30 inch service water header terminus, main discharge tunnel, AOG/NRW service water common header and the main cooling water intake at the intake structure for pH and total residual chlorine.

On July 26, 1990, the inspector reviewed the results of the above sample analysis for the month of July and had no significant observations.

3.5 Monthly Maintenance Observation

The inspector observed performance of the following maintenance on the dates shown:

8/15/90 Replacement of reflash units in control room panel ER-43 (work request No. 61326, job order No. 25195)

8/16/90 Repacking containment spray suction valve V-21-7 (work request No. 750795, job order No. 24209)

The inspector verified that appropriate administrative approvals were obtained, equipment tagout was adequate and properly done, the technicians were following the required procedure, appropriate quality control hold points were instituted and radiological control was adequate. One exception noted during V-21-7 repacking occurred when the stuffing box measurements were not taken after removal of old packing. Procedure A100-GMM-3917.51, Rev. 0, "Installation and Use of Chesterton Packing," requires, in step 4.4.3, that during preinstallation inspection the depth and inside diameter of the stuffing box be measured. GPUN later found that the stuffing box did not have the required depth to hold all three Chesterton packing rings. The installed packing was removed and after an engineering review the carbon bushing was reduced in size to make room for a third packing ring. The inspector concluded this error did not have any safety significance, was corrected, and did not have any other questions.

The involvement of the radiological control technician (RCT) during the valve repacking job was noteworthy. The RCT periodically monitored the radiological conditions and stopped the job when one smear sample showed contamination. The radiological conditions were evaluated and the job was restarted with additional radiological controls. The inspector had no other observations of radiological conditions.

4.0 Engineering and Technical Support

4.1 Emergency Diesel Generator

On July 9, 1990, during a surveillance test, the No. 2 emergency diesel generator did not automatically trip at 350 ± 100 KW and during unloading experienced load oscillations. During reduced voltage start the operators noticed slow engine start and a smell of acid in the diesel cubicle. One dedicated 120 volt DC battery is provided for each diesel generator. Each battery consists of 56 cells and provides power to the starter motors, generator field flashing loads and control power. The diesel generator was declared inoperable and the plant entered a seven-day limiting condition for operation (LCO) as required by the technical specifications.

The licensee's troubleshooting consisted of monitoring diesel battery individual cell voltage during engine start (crank test). The inspector observed the performance of cell voltage during a crank test and discussed the diesel performance with the plant engineer. During this troubleshooting the licensee identified a degraded cell.

The de ded cell was replaced and, after adjustment of the rack switch, the diesel generator was declared operable on July 13, 1990 and the seven-day LCO terminated.

The licensee reviewed the perfor ance of both diesel generators during several past surveillance tests. During an April 17, 1990, surveillance test the No. 1 diesel generator failed to start in the slow roll mode. The cause of the failure was also attributed to a degraded battery cell. During both of these events the emergency start capability of the diesel generators was maintained. The licensee decided to perform a periodic monitoring of battery cell voltage during engine startup and to replace the batteries in both diesel generators before 13R refueling outage. The inspector did not have any other questions.

4.2 Diesel Generator Load Swings

On August 15, 1990, inspectors reviewed with a plant engineer the troubleshooting and corrective action associated with emergency diesel generator No. 2 electrical load swings. During surveillance testing, load swings of approximately 200-300 kw were observed.

Licensee troubleshooting identified the cause of load swings as loosening of a subgovernor case inside Woodward governor actuator (EGB 13C Woodward governor/actuator model No. 8240-762). This loosening caused intermittent erratic output of the governor and subsequent oscillating of diesel generator load during surveillance test. Three screws inside the governor were tightened, the generator load limit and rack switch were readjusted, and the emergency diesel generator was satisfactorily tested. The licensee reviewed the stability of No. 1 emergency diesel generator and concluded it was satisfactory.

The inspector concluded that the loosening of this subgovernor housing is a long term effect that was satisfactorily captured by the licensee's periodic surveillance test program.

5.0 Observation of Physical Security

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

The inspector had no notable observations.

6.0 Safety Assessment/Quality Verification

6.1 Tensiometer used during 1B2 Cable Installation Found out of Calibration

The licensee sent the tensiometer used in 1B2 cable installation to the calibration laboratory after completion of the installation during the 12U-K outage. This tensiometer was found out of tolerance and could not be calibrated. The lab reported that the tensiometer readings were lower than the actual tension in the cable during the test. Since the cable received a maximum pulling tension of 5,000 lb per tensiometer reading, the licensee determined that the manufacturer's specified maximum pull tension of 6,000 lb could have been exceeded during the cable installation. A Material Nonconformance Report and a Deviation Report were written to evaluate the deficiency.

The licensee concluded that no damage occurred to the 1B2 cable due to overtensioning during installation. This was based on physical and electrical tests performed on the cable after installation in the conduit and also on a sample of the cable that was pulled through the conduit. Field tests on the installed cable included 35 KV DC Hypot and a 5KV AC power factor test had acceptable results. A cable sample that was pulled through the conduit was sent to a test lab. This cable did not show any physical damage due to overtensioning upon visual examination. Jacket adhesion test and partial discharge, AC/DC breakdown voltage and power factor tests performed by the lab showed that the cable was in a good condition. A periodic test program for installed 4.16 KV cables is currently being developed for implementation by 13R outage.

A tool calibration deficiency report was prepared for the tensiometer. The tensiometer has been taken out of service and separated with a "not to be used" tag. The tensiometer was on a three-month calibration cycle. The meter was calibrated in March 1990, approximately a month before it was used in the 1B2 cable installation. The licensee indicated that the tensiometer was not used in any other job during this interval. During cable pulling, the tensiometer suffered a strong reaction force when a pulley anchored to a wall was dislodged. The licensee believes this could have damaged the tensiometer and caused it to go out of calibration. The inspector concluded the licensee had established adequacy of the installed 1B2 cable, the nonconforming tensiometer was taken out of service, and the calibration frequency of the tensiometer was adequate. The inspector had no other questions.

6.2 Air Receiver Relief Valve

The inspector reviewed the licensee's critique for the non-safety number 3 air receiver relief valve that prematurely lifted on July 17, 1990, to determine if root cause and corrective actions were adequately identified. The licensee had experienced past failures of Lonergan Company relief valves in the core spray system. The number three air receiver relief valve was replaced with a Lonergan Company Model 11 W 203 series 4800 valve during preventive maintenance. The work order called for a 6800 series valve which was not available at that time. A 4800 series valve was selected by the work crew without engineering review based on its set point (125 psi) which was the same and relief capacity which was somewhat higher but close to what was required. After installation of the new valve, the system was returned to service and number one air receiver was removed from service for similar preventive maintenance. After a 15 hour run, the new valve on number three air receiver lifted several times. The air receiver was isolated after the service/control system low pressure alarm was received. The air system pressure dropped to 68 psi. The licensee bench tested the original relief valve and replaced the newly installed valve with the original valve.

The licensee bench tested the failed valve, and the other 4800 and 6800 series valves. The failed valve lifted at 102 psi. The other 4800 series valve did not lift up to 170 psi. The 6800 series valves demonstrated set points close to their specification. To determine the root cause of the failure, the valve was sent back to the manufacturer. The corrective actions identified in the critique included addition of preinstallation testing requirements and evaluation of this failure for possible common mode with the past failure of Lonergan relief valves installed in the core spray system. Upon inspector's questions, the licensee indicated that warehouse control of replacement parts is being enhanced via a hold tag procedure which would require Plant Engineering review and approval for replacement parts. Plant Procedure 105, Rev. 32, "Control of Maintenance" will be revised to include a similar engineering review of replacement parts. The inspector did not have any other questions.

6.3 Routing of Additional Gas Pipeline Along Highway 9

During May 1990 the New Jersey Gas Company laid a 16 inch diameter natural gas pipeline along the west side of highway #9 which runs in close proximity of the GPUN property containing the Oyster Creek plant. At the south end of this property this line crosses the discharge canal. At the north end of the property the 16 inch line branches off with an additional 16 inch line running by the side of the intake canal and supplies natural gas to the GPUN gas turbines. This gas line was an addition to a 6 inch line which runs by the side of highway #9 along the same route.

The inspectors reviewed the licensee's safety evaluation for the additional natural gas pipeline routed along highway #9. This safety evaluation assessed the risk of possible fire or explosion resulting from a gas line leak and its effect on safety related structures and equipment in the plant and on control room habitability. The safety evaluation determined that the gas line does not pose a threat to the safe operation of the plant.

Regional personnel performed a bounding analysis of the [illegible]er Creek pipeline installation with previously acceptable install[illegible]. The general methodology that was applied is contained in NUREG-C[illegible] Hartsville Safety Evaluation Report (SER). That SER referenced [illegible]cs Research Incorporated study funded by Tennessee Valley Authority.

Based on the above reviews, the inspectors concluded that the licensee's safety evaluation conclusion was sound; that is, no undue risk was associated with the pipelines near Oyster Creek.

6.4 Skin Contamination and Radiological Intake during Repacking of Shutdown Cooling System Valve

The inspector reviewed the licensee's critique of an inc[illegible] on July 2, 1990, during which two mechanics received skin contamination and had a radiological intake of 9.39 and 2.69 mpc hr respectively. The inspectors' review of the event is contained in report 90-11, section 2.1. The incident happened during repacking shutdown cooling system valve V-17-56. Due to high contamination levels in the room and high contact doses involved, an ALARA review was performed. The ALARA review required use of a HEPA ventilation unit in the work area. Use of respirators during installation of the new packing was left up to the GRCS involved and was decided not to be necessary. The radiological controls technician (RCT) involved with the job was to verify that the HEPA ventilation was generating an adequate capture velocity.

Increased airborne activity during installation of the new packing, together with a HEPA unit that was not effective was identified as a possible cause of intake. The use of full face respirators instead of face shields could have prevented this intake and skin contamination on the face. The licensee prepared a lessons learned document which discussed better radiological practices for minimizing levels of airborne contamination and use of respirators for contamination control. The lessons learned document was distributed to the GRCSs and RCTs to be stressed during prejob briefings and discussions.

Site Services Department feedback to the critique indicated that future work will be performed in close communication with Radiological Controls and craft personnel. The inspector concluded the licensee had identified the cause of radiological intake and skin contamination, and the necessary corrective actions. The inspector did not have any other questions.

6.5 Use of Teflon Tape in Reactor Building

Inspectors questioned the use of Teflon tape as thread sealant material in the control rod drive hydraulic system scram air header. Licensee evaluation considered the following:

- General Electric Company specifies the use of teflon tape on the Hydraulic Control Units (GEG-30702);

- Teflon tape is not recommended in areas that come into contact with reactor water fluids (NEDE 31295P); and,

- Teflon tape is not recommended in high temperature or high radiation areas due to breakdown of material.

Licensee evaluation concluded teflon tape is an acceptable material for use as a thread sealant in the CRD air system where low temperature, low radiation, and non-reactor water medium exist.

Inspectors reviewed the licensee evaluation and concluded it was thorough and complete.

6.6 Feedwater Flow Calibration Error

During routine reviews of plant thermal performance, the licensee identified that during 1987 an error had existed in the feedwater flow calibration equation. In 1987 this error was identified and corrected by a procedure change. Current review identified that it was possible that the reactor plant had been operating above its licensed limit of 1930 Mwt and that no apparent review for reportability had occurred. Current licensee review concluded that this condition is reportable. The licensee is in the process of generating a licensee event report. Possible operation of the plant above its licensed limit, 1930 Mwt, will remain unresolved pending NRC review of the licensee evaluation of this condition (UNR 50-219/90-12-01).

6.7 Spill Resulting from Filling and Venting of Shutdown Cooling System

On June 26, 1990, while filling and venting of shutdown cooling (SDC) system prior to returning to service, a spill occurred in the reactor building. During filling and venting, water was released into a hub drain in the shutdown cooling heat exchanger room at elevation 51 ft of the reactor building. Water came out of a floor drain at the northwest corner at the 51 ft elevation of the reactor building and resulted in the spill.

The inspector reviewed the licensee's critique of the spill to determine if an appropriate root cause and the necessary corrective actions were identified. The root cause of the spill was identified to be a clogged floor drain system. The licensee demonstrated that adding water in the hub drain in the SDC heat exchanger room resulted in a spill from the floor drain in the northwest corner at the 51 ft elevation of the reactor building.

A wide spectrum of deficiencies identified in the critique include no periodic requirements to inspect and clean the floor drains, long term packing leak on SDC system valve V-17-56, procedure for filling and venting which did not provide clear guidance, and the sequence (one loop at a time versus all three loops simultaneously) was not strictly followed. The critique also indicated that the packing leak on V-17-56 caused a spill on April 23, 1990; however, no deviation reports were written on the degraded condition of the valve.

The inspector reviewed the corrective actions with the licensee. The clogged floor drain was cleaned. Long term corrective actions included periodically cleaning the floor drain system and revising the procedure for filling and venting the SDC system. A memorandum was prepared by and distributed to the radiological controls personnel on the need to submit deviation reports. The inspector concluded there was no safety significance to filling one SDC loop at a time as opposed to filling all three together. The inspector did not have any other questions.

6.8 Conclusions

Inspector review of licensee critiques and corrective action show that the licensee is conducting in-depth and thorough reviews. Important aspects are identified and appropriate actions are specified for correction or enhancement. Engineer inquisitiveness uncovered the fact that an old (1987) error was not adequately reviewed for reportability. Proactive safety reviews for the gas line installation ensured plant safety.

7.0 Inspection Hours Summary

Inspection consisted of 250 direct inspection hours; 36 of these direct inspection hours were performed during backshift periods, and 8 of these hours were deep backshift hours.

8.0 Meetings and Unresolved Items

8.1 Commissioner's Visit

On July 31, 1990, Commissioner James R. Curtiss visited Oyster Creek site. The Commissioner met with licensee management and the resident inspectors, toured the plant and attended the licensee's Plan of the Day meeting. The licensee's presentation is contained in Attachment II.

8.2 Preliminary Inspection Findings

A verbal summary of preliminary findings was provided to the senior licensee management at the conclusion of this inspection. During the inspection, licensee management was periodically notified verbally of the preliminary findings by the resident inspectors. No written inspection material was provided to the licensee during the inspection. No proprietary information is included in this report.

8.3 Unresolved Items

Unresolved items are matters for which more information is required in order to ascertain whether they are acceptable, violations or deviations. An unresolved item is discussed in section 6.6 of this report.

Attachment I

Sequence of Events of August 6, 1990

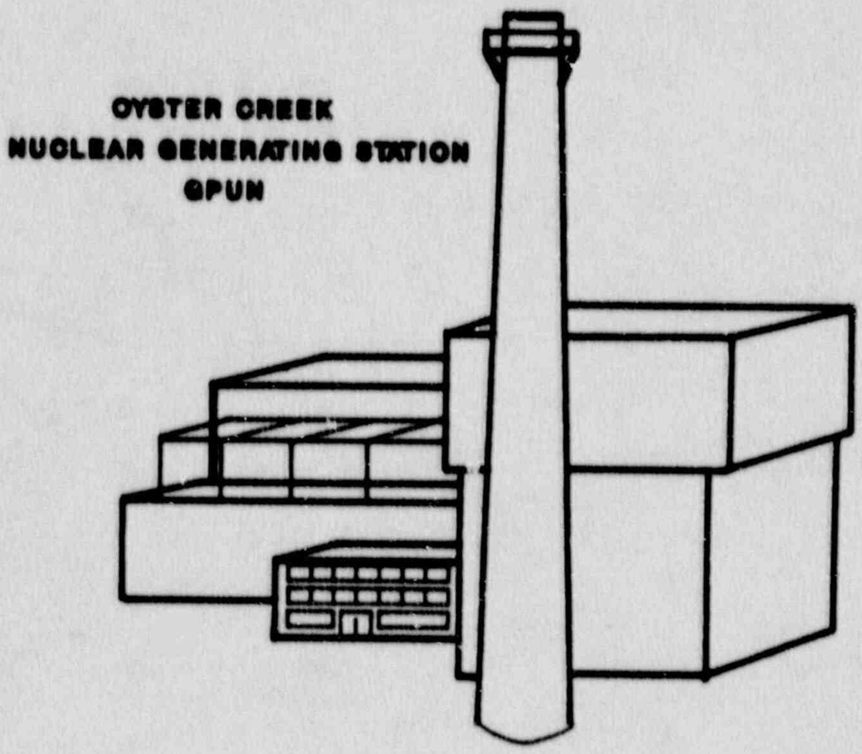
Time	Source	Description
12:57:34	PTRG	Valve V-21-5 indicates closed
12:57:39	PTRG	Valve V-21-5 deenergized (not fully closed)
14:37:02	SAR	Containment spray system II autostart
14:37:45	SAR	Containment spray system II pump 51-C starts
14:39	P1 Comp	Unidentified leak rate recorder goes upscale (10 gpm)
14:39:56	SAR	First of several drywell temp hi alarms
14:40:15	SAR	Pump 51-C trips (I&C surveillance step)
14:40:45	SAR	Drywell pressure hi/lo alarm on high pressure
14:42	Estimated	Commenced venting drywell and torus
14:43	P1 Comp	First pumpdown of 108 sum started
14:50:04	SAR	Drywell pres high alarm resets
14:59	Estimated	Secured venting of drywell and torus
15:00	Samp Sheet	Drywell sump sample #1 obtained
15:09	P1 Comp	1-8 sump pump secures
15:13	P1 Comp	1-8 sump starts second pumpdown
15:13:04	SAR	Drywell sump hi leak alarm
15:14:17	SAR	Drywell sump hi leak rate alarm reset
15:15	GSS Log	Unusual event declared and shutdown ordered
15:18	CRO Log	Verified both sump pumps 1-8A and B running, integrators. Attempted to manually close V-21-5, no movement
15:31	Samp Sheet	Drywell sample #1 count completed
15:41:59	SAR	Drywell sump hi leak rate alarm
15:43:31	SAR	Drywell sump hi leak rate alarm resets

Time	Source	Description
16:15	Samp Sheet	Drywell sump sample #2 obtained
16:37	Samp Sheet	Drywell sump sample #2 count complete
17:50	CRO Log	Secured from Unusual Event
18:10	CRO Log	Terminated plant shutdown
19:01	Samp Sheet	Drywell ring header sample taken
19:45	Samp Sheet	First count completed drywell ring header
20:06	Samp Sheet	Second count completed drywell ring header

DEFINITIONS OF ABBREVIATIONS USED

PTRG	Plant Trip Review Group
SAR	Sequence of Alarm Recorder
Pl Comp	Plant Computer
Samp Sheet	Sample Sheet
GSS log	Group Shift Supervisor Log
CRO log	Control Room Operator Log

ATTACHMENT II
Report 50-219/90-12



GENERAL PUBLIC UTILITIES NUCLEAR
COMMISSIONER JAMES CURTISS VISIT
JULY 31, 1990

GPUN ATTENDEES

PLANT OPERATIONS DIRECTOR	R.J. BARRETT
DEPUTY DIRECTOR OC	J.J. BARTON
LICENSING MANAGER OC	G.W. BUSCH
DIRECTOR TECHNICAL FUNCTIONS OC	G.R. CAPODANNO
PRESIDENT	P.R. CLARK
MANAGER QA MOD/OPS OC	R.F. FENTI
MANAGER PLANT TRAINING	J.D. KOWALSKI
PLANT MAINTENANCE DIRECTOR	L.L. LAMMERS
PLANT ENGINEERING DIRECTOR	A.H. RONE
RADIOLOGICAL CONTROLS DIRECTOR OC	M.J. SLOBODIEN
COMMUNICATIONS REPRESENTATIVE	S.M. TOD[illegible]

VISIT TO OCNGS BY NRC COMMISSIONER JAMES CURTISS

AGENDA

JULY 31, 1990

8:00 - 8:45	ATTENDANCE AT PLAN OF THE DAY MEETING
8:45 - 11:00	TOUR OF THE PLANT
11:00 - 12:00	WORKING LUNCH WITH PRESENTATIONS BY: P. CLARK, PRESIDENT GPUN J. BARTON, DEPUTY DIRECTOR OC J. KOWALSKI, MANAGER PLANT TRAINING M. SLOBODIEN, RADIOLOGICAL CONTROLS DIRECTOR L. LAMMERS, PLANT MAINTENANCE DIRECTOR
12:00 - 1:30	PERSONAL INTERVIEWS 12:00 - 12:30 R. BARRETT 12:30 - 1:00 M. SLOBODIEN 1:00 - 1:30 L. LAMMERS
1:30 - 1:45	EXIT WITH SENIOR MANAGEMENT

TOUR ROUTE

CONTROL ROOM

ENTER RCA (ACROSS FROM CONTROL ROOM)

TURBINE DECK

TURBINE BLDG. BASEMENT SOUTH (VIA STAIRWELL)

COND. DEMIN. ROOM (VIA STAIRWELL BY MFP ROOM)

FEEDPUMP ROOM

CRD PUMP ROOM

RX BLDG ELEVATION 23' - N.E. CORNER ROOM AREA 19'

RCA YARD AREA - NRW & AOG BLDGS.

REACTOR BLDG 23'

ELEVATOR TO RX BLDG. 119' - REFUEL BRIDGE

REACTOR BLDG. 95' (VIA STAIRWELL)

RX BLDG. 75' (VIA N.W. STAIRWELL) - CRD REBUILD ROOM

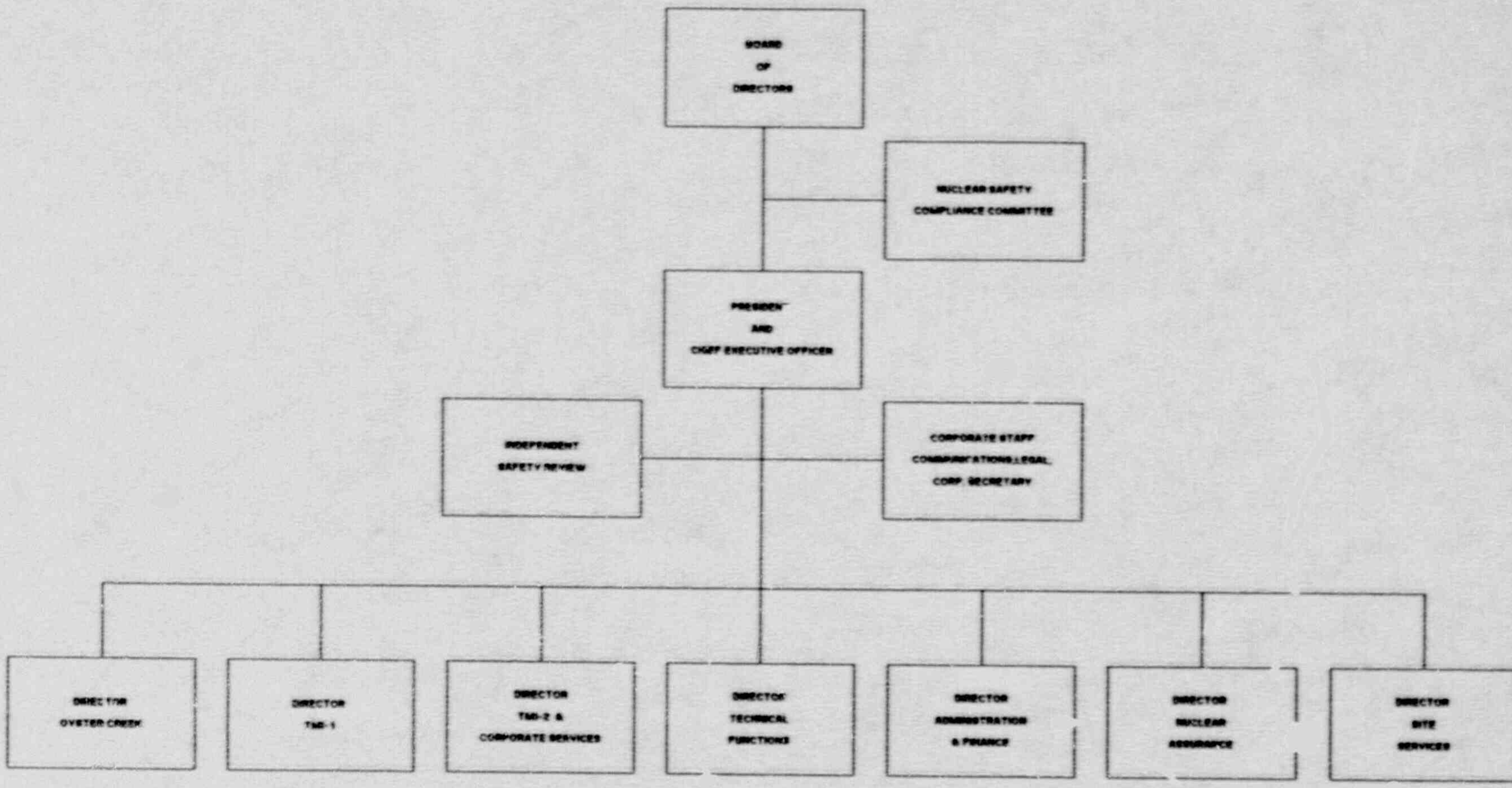
RX BLDG. 51' (VIA S.E. STAIRWELL)

EXIT RCA (@ MAC)

INTAKE/DILUTION STRUCTURES

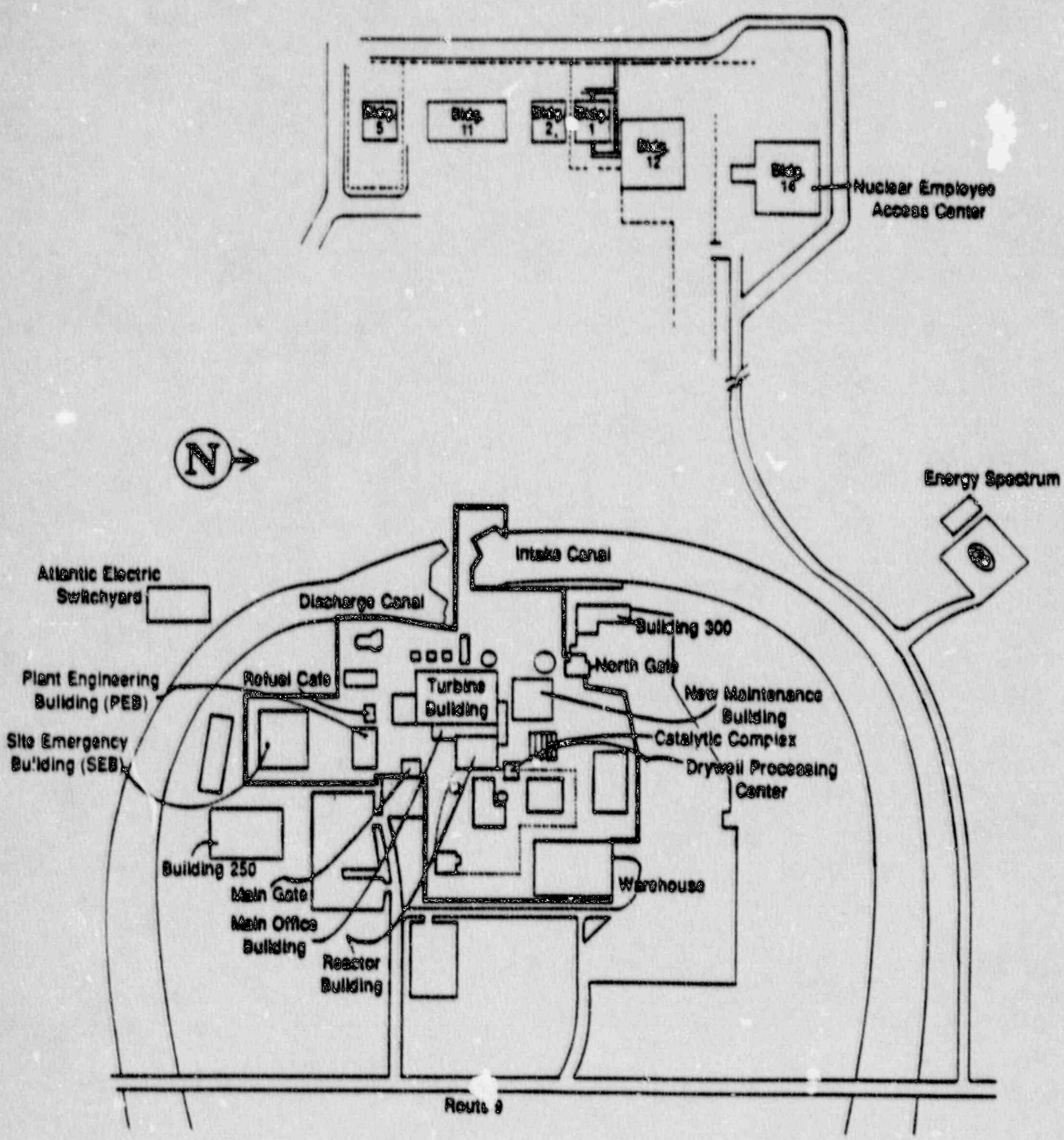
TSC

GPU NUCLEAR CORPORATION

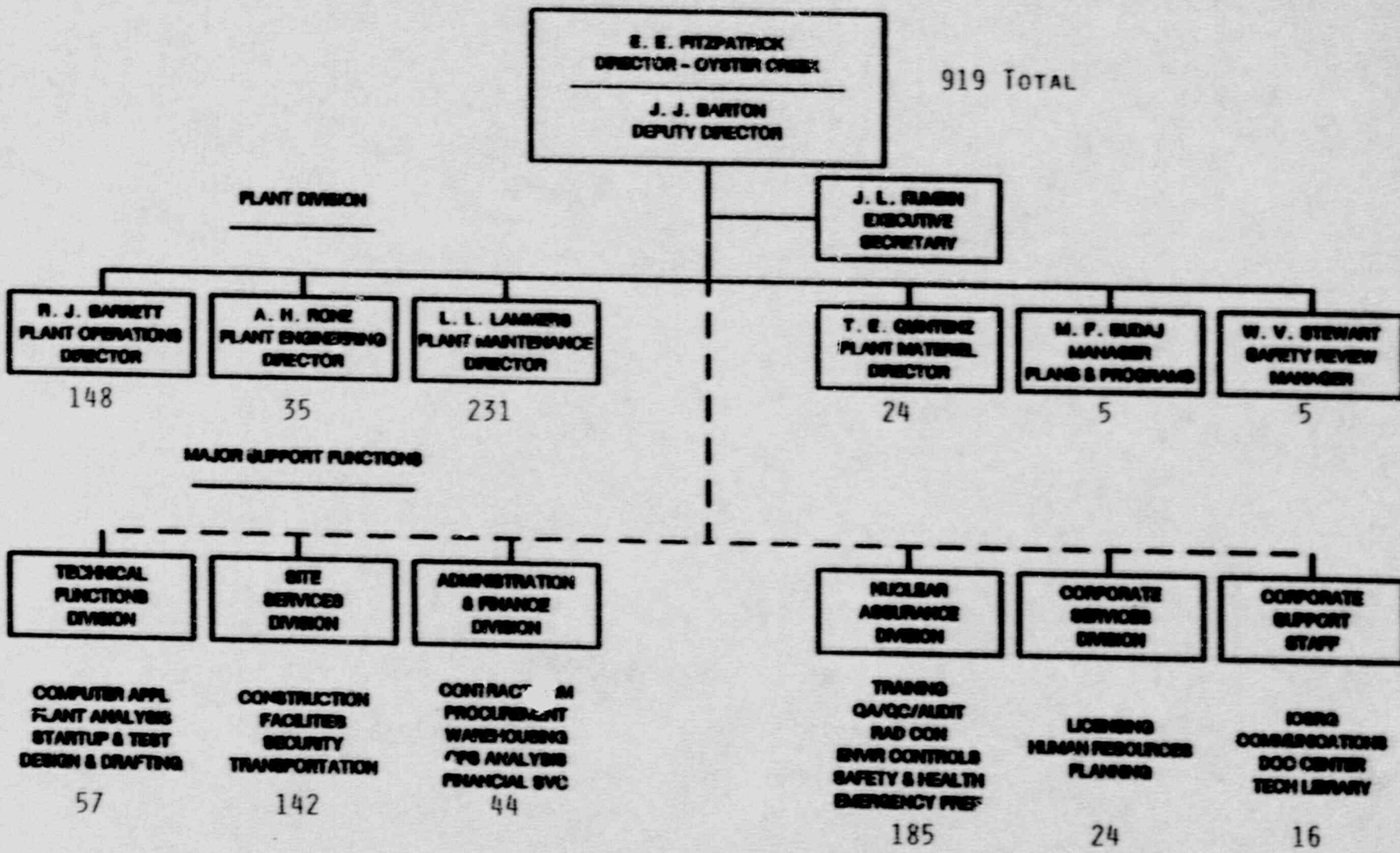


OYSTER CREEK STATION

OYSTER CREEK SITE MAP



OYSTER CREEK SITE ORGANIZATION



ON-SHIFT MANNING

OPERATIONS	11
MAINTENANCE	7
CHEMISTRY	1
RADIOLOGICAL CONTROLS	3
STA	1

EXPERIENCE/EDUCATION

SENIOR MANAGEMENT PERSONNEL

NAME	EXPERIENCE (YEARS)	EDUCATION BACHELOR	MASTER	MASTER+
E. E. FITZPATRICK	23	X	X	X
J. BARTON	33	X		
R. BARRETT	30	X	X	
P. SCALLON (SRO)	21	X		
R. BROWN (SRO)	14	X		
K. MULLIGAN (SRO)	10	X	X	
R. HILLMAN	16	X	X	
A. RONE (SRO)*	20	X		
T. DEMPSEY	20	X	X	
J. DeBLASIO	19			
D. RANFT	17	X		X
L. LAMMERS	30	X	X	
G. TRUE	29			
P. FISCHLER	23			
W. MUEHLEISEN	26			
R. BLOUCH	14	X	X	
W. QUINLAN	26			
W. STEWART (SRO)	21	X		
M. SLOBODIEN	19	X	X	
D. TUTTLE	34			
M. BUDAJ	16	X		
T. QUINTENZ	20	X	X	
J. KOWALSKI (SRO)*	15	X	X	
J. FREW	29	X		
W. BEHRLE	23	X		
G. CAPODANNO	23	X	X	
R. FENTI	25			
E. ROESSLER	31	X		
N. CHRISSOTIMOS	16	X	X	
TOTAL	643	21	12	2

* PREVIOUSLY HELD SRO LICENSE FOR OYSTER CREEK

PLANT STATUS

POWER LEVEL	100%
CONTINUOUS DAYS ON LINE	26
ANNUNCIATOR STATUS	BLACK

OYSTER CREEK PERFORMANCE
CYCLE 12 (BEGAN 5/11/89)

CAPACITY FACTOR FOR CYCLE 12: 69.04 %

DATA AS OF JULY 16,1990
DATA BASED ON GENERATOR BREAKER
OPEN / CLOSED TIMES

PLANT PERFORMANCE TRENDS

1987 - 1989

NRC VIOLATIONS	**DOWN 40%**
TS VIOLATIONS	**DOWN 70%**
SCRAMS (ALL)	**SAME**
TOTAL LER'S	**DOWN 44%**
- EQUIPMENT FAILURES	**DOWN 40%**
- PROCEDURES	**DOWN 67%**
- PERSONNEL ERROR	**DOWN 73%**
INDUSTRIAL SAFETY	
- LOST TIME ACCIDENT RATE	**DOWN 81%**

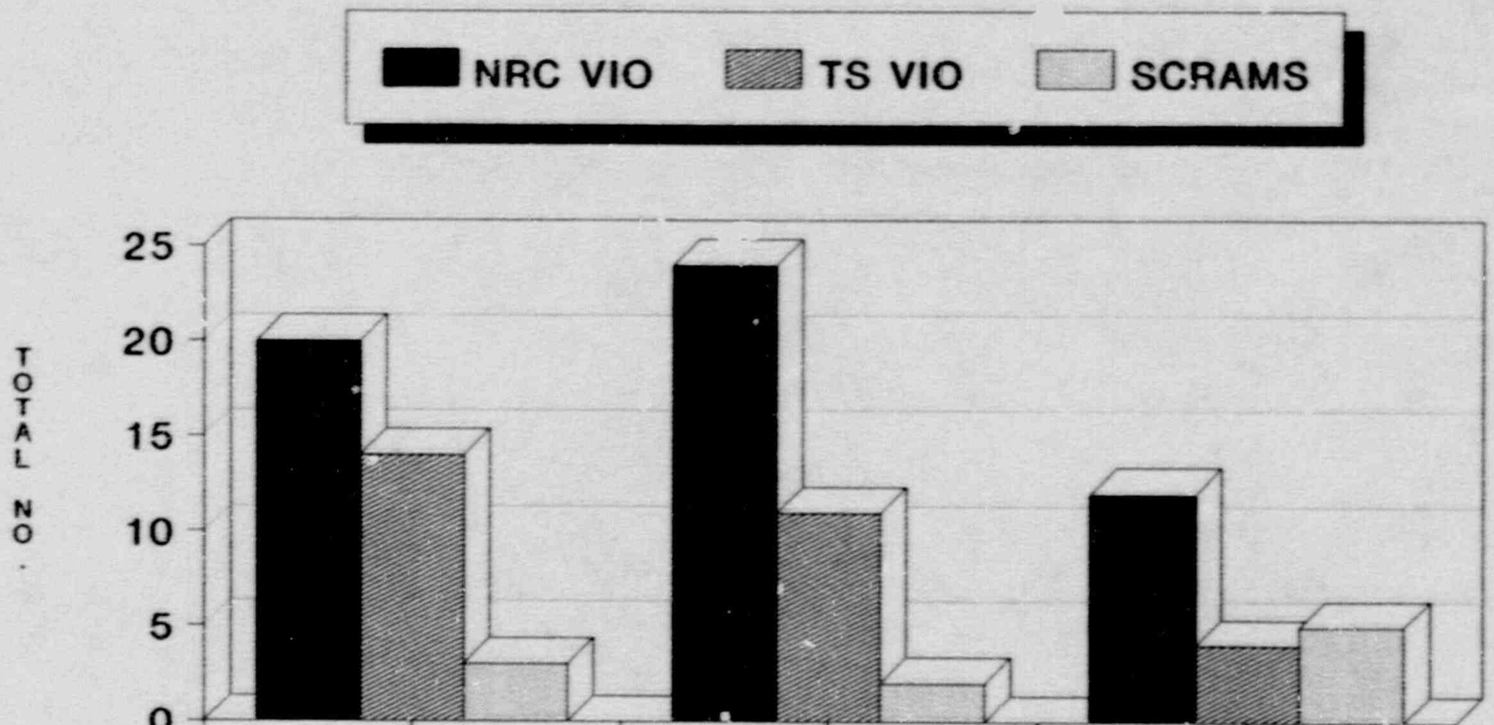
1990

PLANT PERFORMANCE

(THRU 6/90)

°	AOG AVAILABILITY	APPROXIMATELY 98%
°	REACTOR WATER CONDUCTIVITY	AMONG TOP BWR'S
°	TIME AUX SYSTEMS CHEM OOS	0
°	CHEMISTRY PERFORMANCE INDEX	AMONG TOP BWR'S
°	DEVIATION REPORTS	USAGE UP 100% VS 8/89
°	SHORT FORMS OUTSTANDING	IMPROVED 12%
°	WORK REQUESTS OVER 90 DAYS	DOWN 37%
°	UNPLANNED AUTO SCRAMS	1
°	UNPLANNED SAFETY SYSTEM ACTUATIONS	0
°	LIQUID RADWASTE	100% RETURN
°	SOLID RADWASTE	PREDICT ABOUT 27% BETTER THAN INPO GOAL

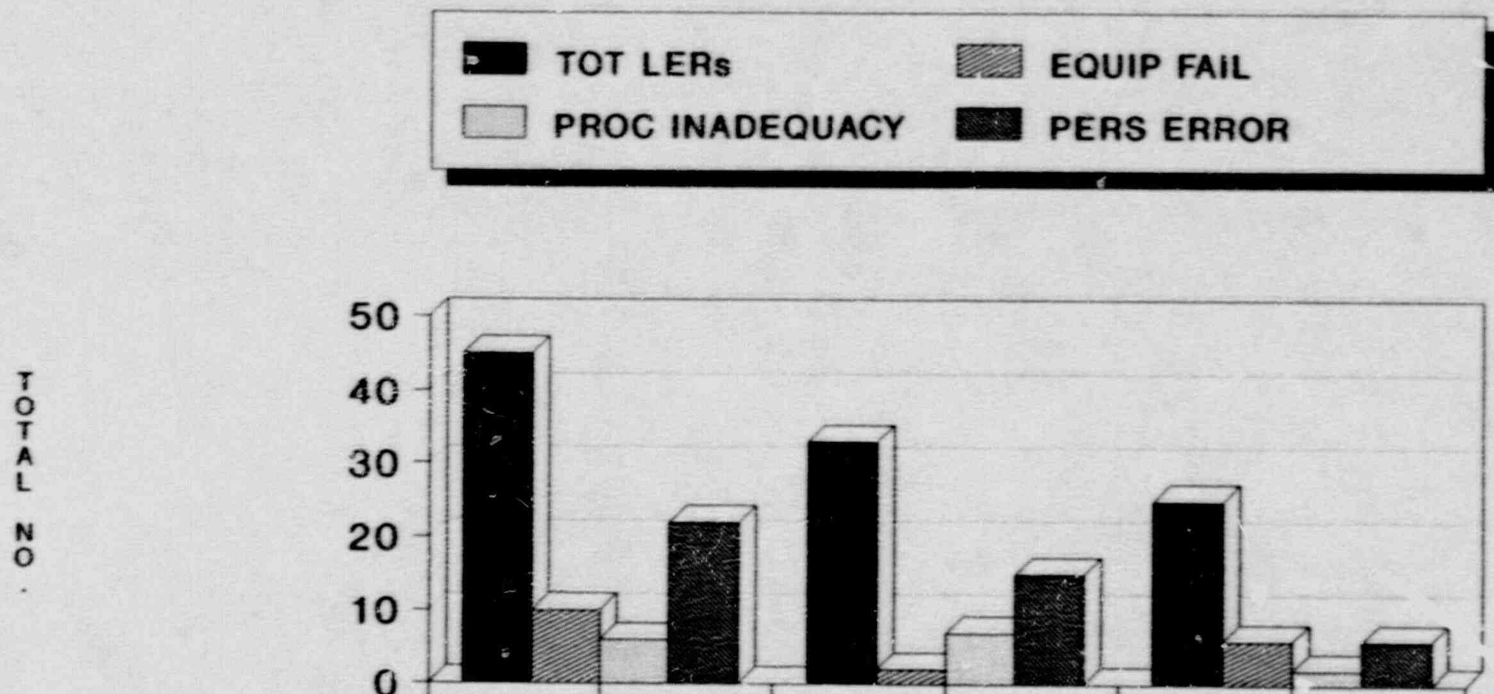
YEARLY STATISTICS
1987 - 1989



	1987	1988	1989
NRC VIO	20	24	12
TS VIO	14	11	4
SCRAMS	3	2	5 •

• TWO DUE TO TRANSFORMER FAILURES

LER STATISTICS
1987 - 1989

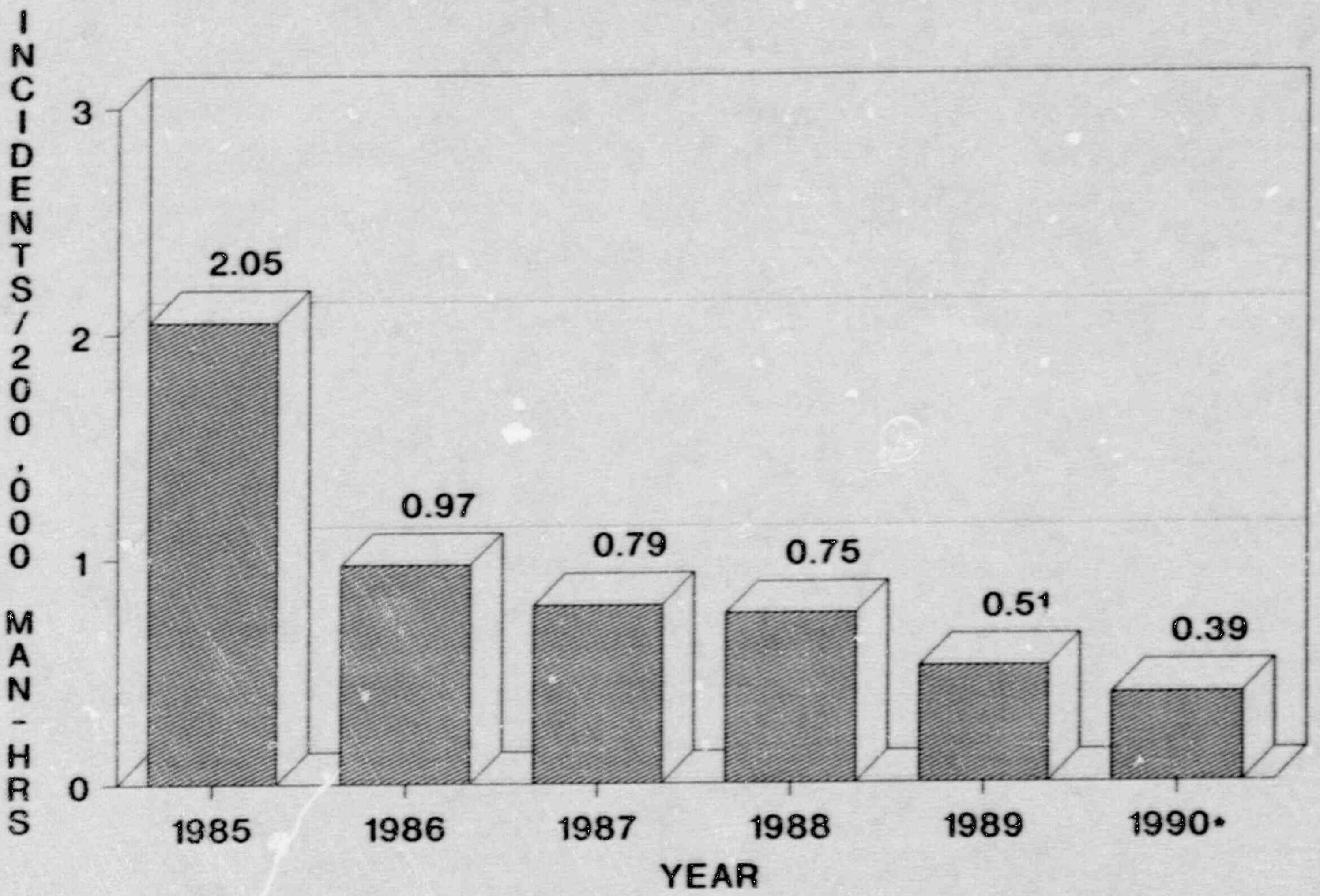


	1987	1988	1989
TOT LERs	45	33	25 *
EQUIP FAIL	10	2	6
PROC INADEQUACY	6	7	2
PERS ERROR	22	15	6

* THREE RPTS SUBMITTED IN 1990

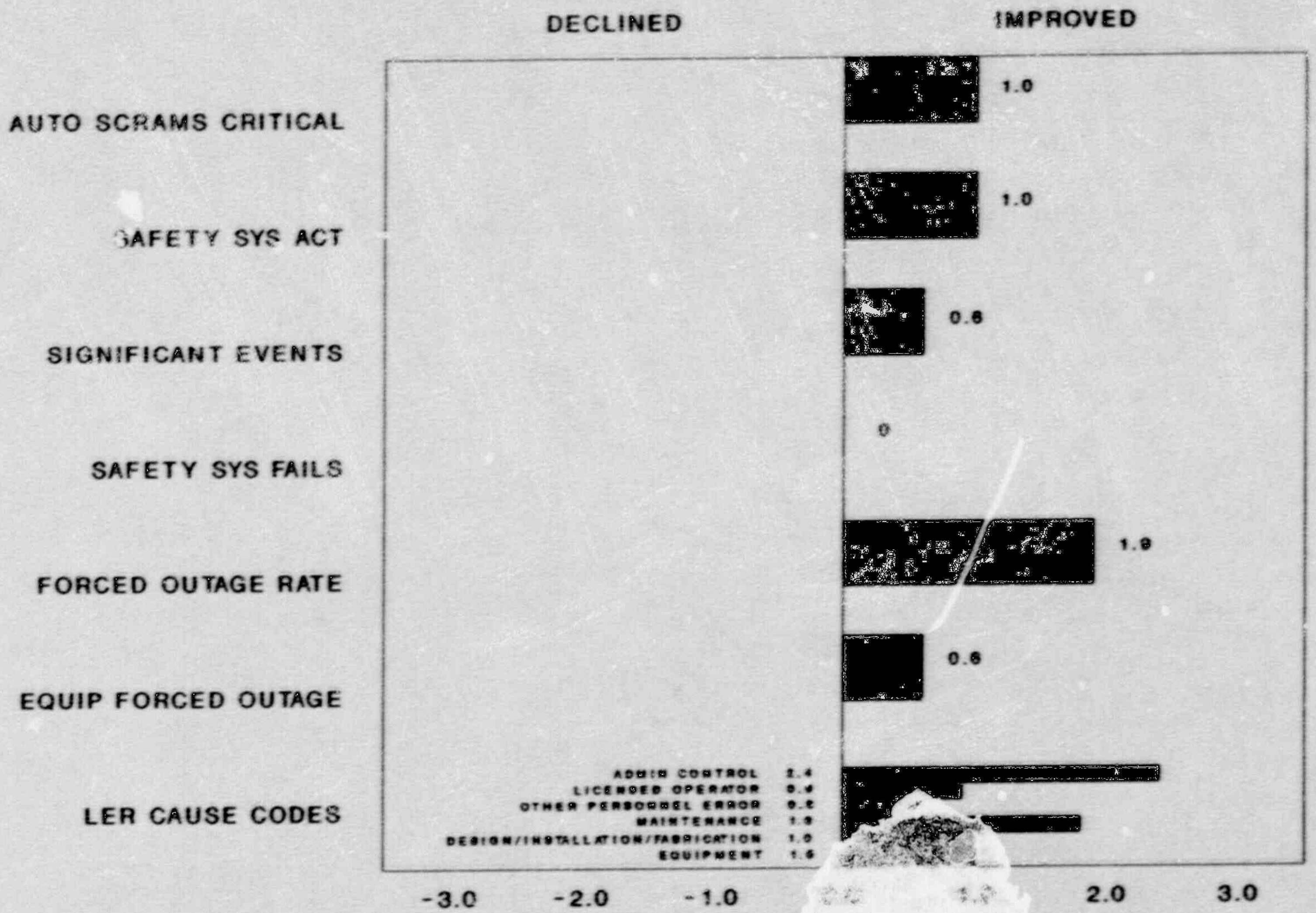
OYSTER CREEK

LOST-TIME ACCIDENT RATE



* AS OF 6/30/90

NRC PERFORMANCE INDICATORS
OYSTER CREEK (1990)



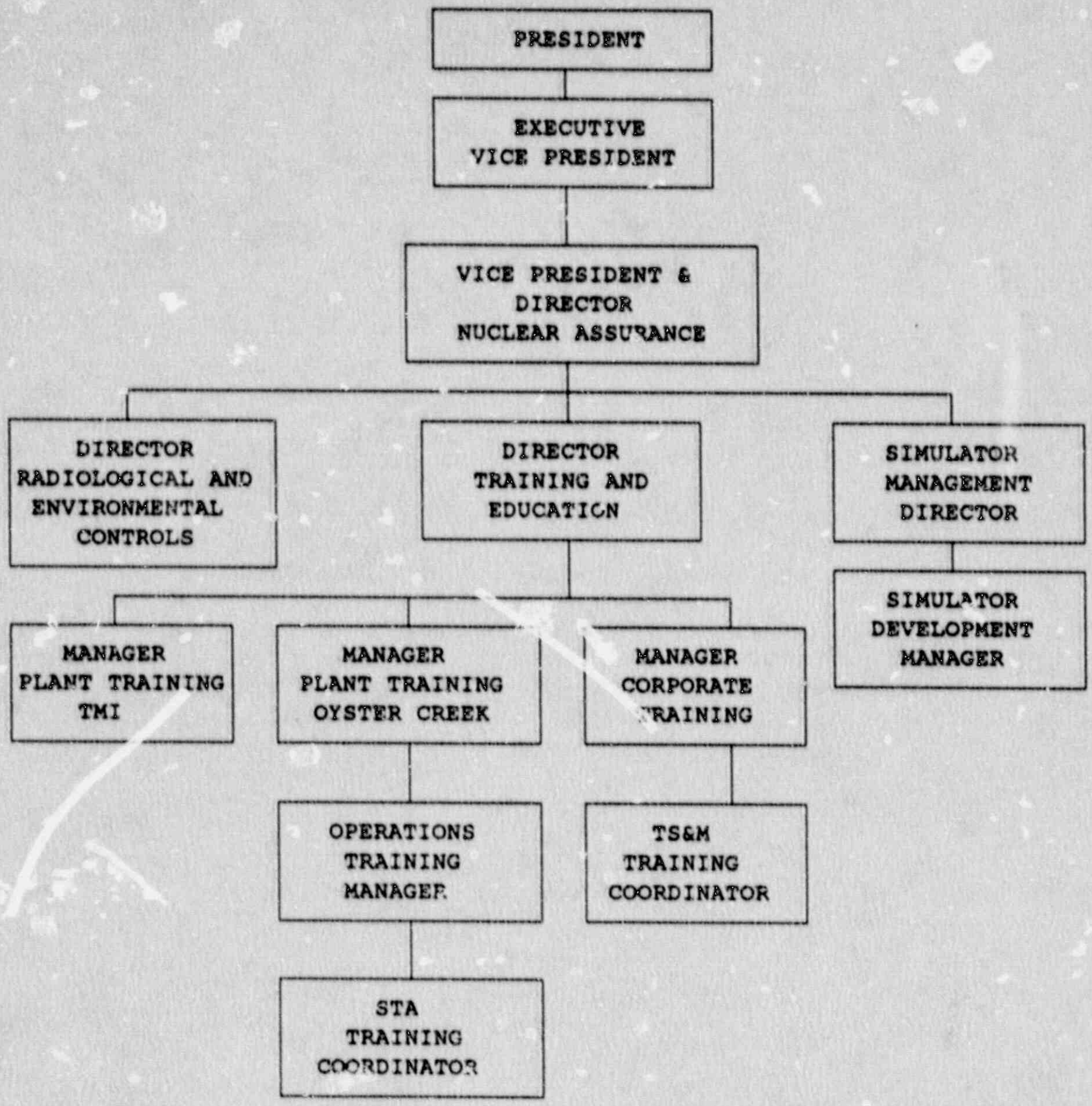
DEVIATIONS FROM [illegible] MEANS
(MEASURED IN [illegible]S)

FIRST QTR (SEC QTR NOT AVAILABLE)

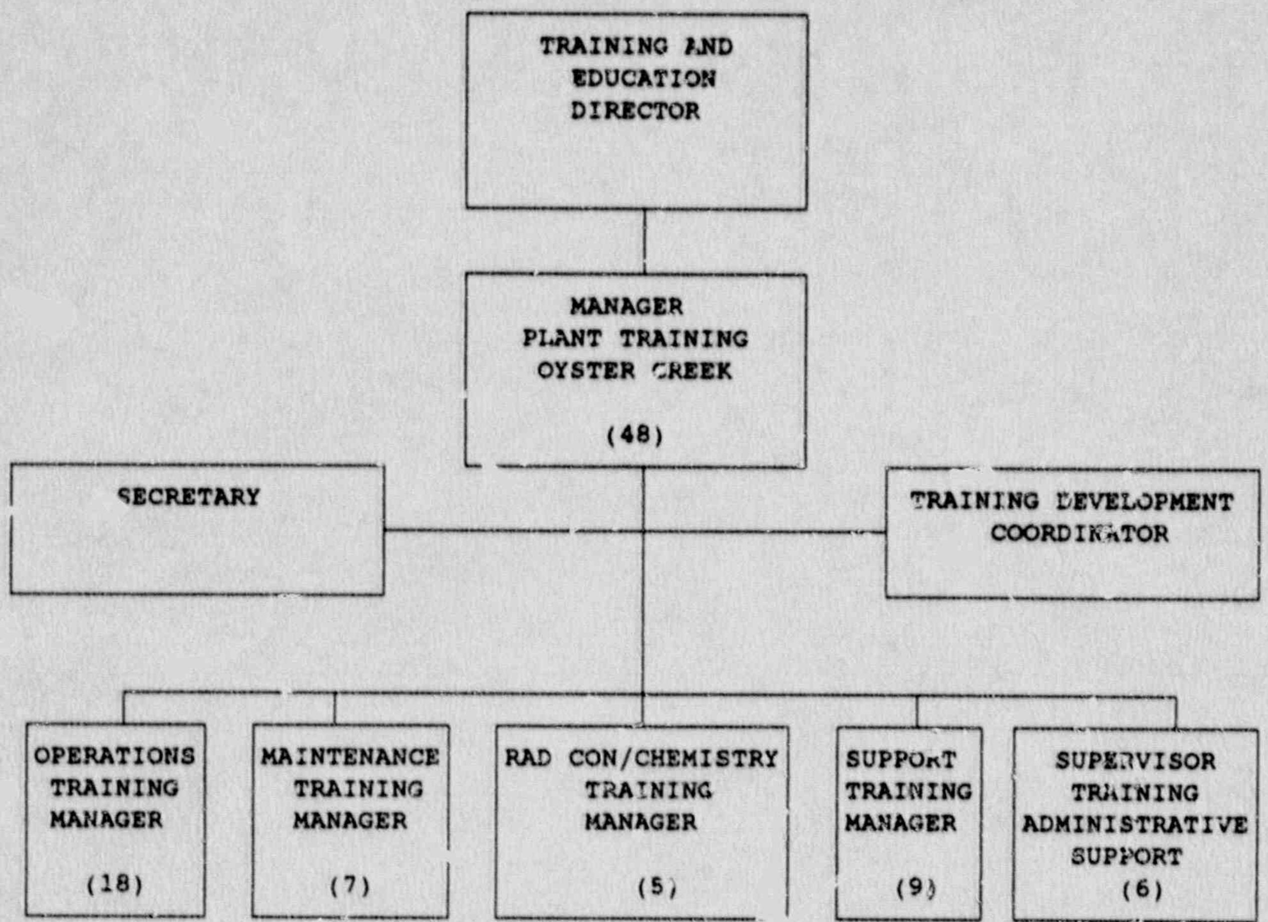
SPECIFIC IMPROVEMENT AREAS

- TRAINING
- RADIOLOGICAL CONTROLS
- MATERIEL CONDITION

OYSTER CREEK TRAINING DEPARTMENT


PRESIDENT
EXECUTIVE VICE PRESIDENT
VICE PRESIDENT & DIRECTOR NUCLEAR ASSURANCE
DIRECTOR RADIOLOGICAL AND ENVIRONMENTAL CONTROLS
DIRECTOR TRAINING AND EDUCATION
SIMULATOR MANAGEMENT DIRECTOR
MANAGER PLANT TRAINING TMI
MANAGER PLANT TRAINING OYSTER CREEK
MANAGER CORPORATE TRAINING
SIMULATOR DEVELOPMENT MANAGER
OPERATIONS TRAINING MANAGER
TS&M TRAINING COORDINATOR
STA TRAINING COORDINATOR

OYSTER CREEK TRAINING



(#) NUMBER OF PERSONNEL

IMPROVEMENTS IN TRAINING FACILITIES AND EQUIPMENT

ENVIRONMENTAL CHAMBER USED IN ADVANCED RADIATION WORK TECHNIQUES ($40,000 IN 1988)

NEW SECURITY FORCE TARGET RANGE ($110,000 IN 1989)

USE OF RADIO CONTROLLED DOSE RATE METERS IN ADVANCED RADIATION WORK TECHNIQUES AND GENERAL EMPLOYEE TRAINING PROGRAMS ($15,000 IN 1989)

FIRE PROTECTION TRAINING BURN PAD ($15,000 IN 1989)

NEW MECHANICAL MAINTENANCE TRAINING BUILDING ALONG WITH MOCK-UPS AND TRAINING AIDS (OVER $1,000,000 1987 - 1990)

INCREASE IN I&C AND ELECTRICAL MOCK-UPS AND TRAINING AIDS ($200,000 1987 - 1990)

ADVANCED RADIATION WORK TECHNIQUES SCENARIO CONSTRUCTION ($100,000 IN 1990)

OPERATOR TRAINING

- INPO ACCREDITATION RENEWAL VISITS, MARCH 1990 (EXPECT ACCREDITATION RENEWAL)

- NRC REQUAL EXAM, APRIL-JUNE 1990 (SATISFACTORY)

- NUREG 1220 AUDIT, JUNE 1990 (REPORT NOT ISSUED)

INPO REVIEW RESULTS

OVERALL RESULTS - EXPECT ACCREDITATION RENEWAL

- ALL ELEVEN (11) PROGRAMS SATISFACTORY
- TWO (2) MINOR CONCERNS
- ACCREDITING BOARD REVIEW IN AUGUST & SEPTEMBER

IDENTIFIED STRENGTH

- LINE MANAGEMENT INVOLVEMENT IN ALL PROGRAMS

NRC REQUAL EXAM

OVERALL RESULTS - PROGRAM RATED SATISFACTORY

- THREE WRITTEN EXAM FAILURES
- TWO JPM FAILURES
- ZERO SIMULATOR FAILURES ON NON-REPLICA SIMULATOR

ISSUE

- WRITTEN EXAM BANK QUALITY (PARTIAL CREDIT, AMBIGUOUS QUESTIONS)
- JOB PERFORMANCE MEASURE (JPM) FORMAT AND ADMINISTRATION (PROCEDURAL ADHERENCE VERSUS RECOVERABLE ERROR)
- SIMULATOR SCENARIO LEVEL OF DETAIL

CORRECTIVE ACTIONS

ALL ACTIONS HAVE BEEN DEFINED, ACCEPTED BY NRC AND ARE BEING IMPLEMENTED

NUREG 1220 AUD!T

ISSUES

- TASK ANALYSIS NEEDS UPDATING (SELF-IDENTIFIED)

ISSUE

- EXAM GRADING I.,SUES - (2 OF 42 OPERATORS REGRADED AS FAILURES)

PROMPT CORRECT!VE ACTIONS WERE TAKEN

GPUN CONCERNS ON REQUAL EXAM

° INCONSISTENCIES IN INTERPRETATION AND APPLICATION OF EXAMINER STANDARD (ES-601).

- INCONSISTENT FROM PLANT TO PLANT
  ° DEPENDENT ON MAKE-UP OF EVALUATION TEAM

  ° NUMBER OF CRITICAL TASKS PER SIMULATOR SCENARIO

- DEVIATION FROM ES-601
  ° ASSIGNMENT OF CREW POSITIONS AT SIMULATOR

° RELATIONSHIP BETWEEN NRC AND UTILITY EVALUATORS
- BEING ADDRESSED WITH NRC STAFF

° REQUAL EXAM WITHOUT REPLICA SIMULATOR

RADIOLOGICAL CONTROLS

RADIOLOGICAL CONTROLS

- IDENTIFYING/UNDERSTANDING OF PROBLEMS
- IMPROVEMENT PROGRAM
- PROGRESS

RADIOLOGICAL CONTROLS

IDENTIFICATION & UNDERSTANDING PROBLEMS

- GPUN SELF ASSESSMENTS
- INPO ROUTINE EVALUATIONS
- INPO ASSISTANCE FOR CONTAMINATION CONTROL
- THIRD PARTY EVALUATION
- NRC INSPECTIONS
  - ROUTINE
  - HEALTH PHYSICS APPRAISAL
  - SALP

RADIOLOGICAL CONTROLS

IMPROVEMENT PROGRAM

- DOSE REDUCTION
  - CHEMICAL DECON OF RECIRCULATION LOOPS & CLEAN UP SYSTEM
  - DRAIN SYSTEM FLUSHING
  - FUEL STORAGE POOL CLEAN-UP PROJECT
  - SHIELDING
  - SYSTEM FLUSHING
  - POWER REDUCTIONS FOR CONDENSER BAY WORK WHILE OPERATING
  - REMOTE TECHNOLOGCY

- PLANNING & EXECUTION OF WORK
  - GMS-2 SYSTEMS
  - GREATER DETAIL IN WORK PLANNING
  - RADIOLOGICAL ENGINEERING DIRECT INVOLVEMENT IN
  - LONG LEAD WORK PLANNING (13R OUTAGE)

- DECONTAMINATION & CONTAMINATION CONTROL
  - MATRIX PLAN
  - DEDICATED STAFF
  - EXTENSIVE PAINTING PROGRAM
  - CHANGE IN BASIC PROTECTIVE CLOTHING PRESCRIPTION

RADIOLOGICAL CONTROLS

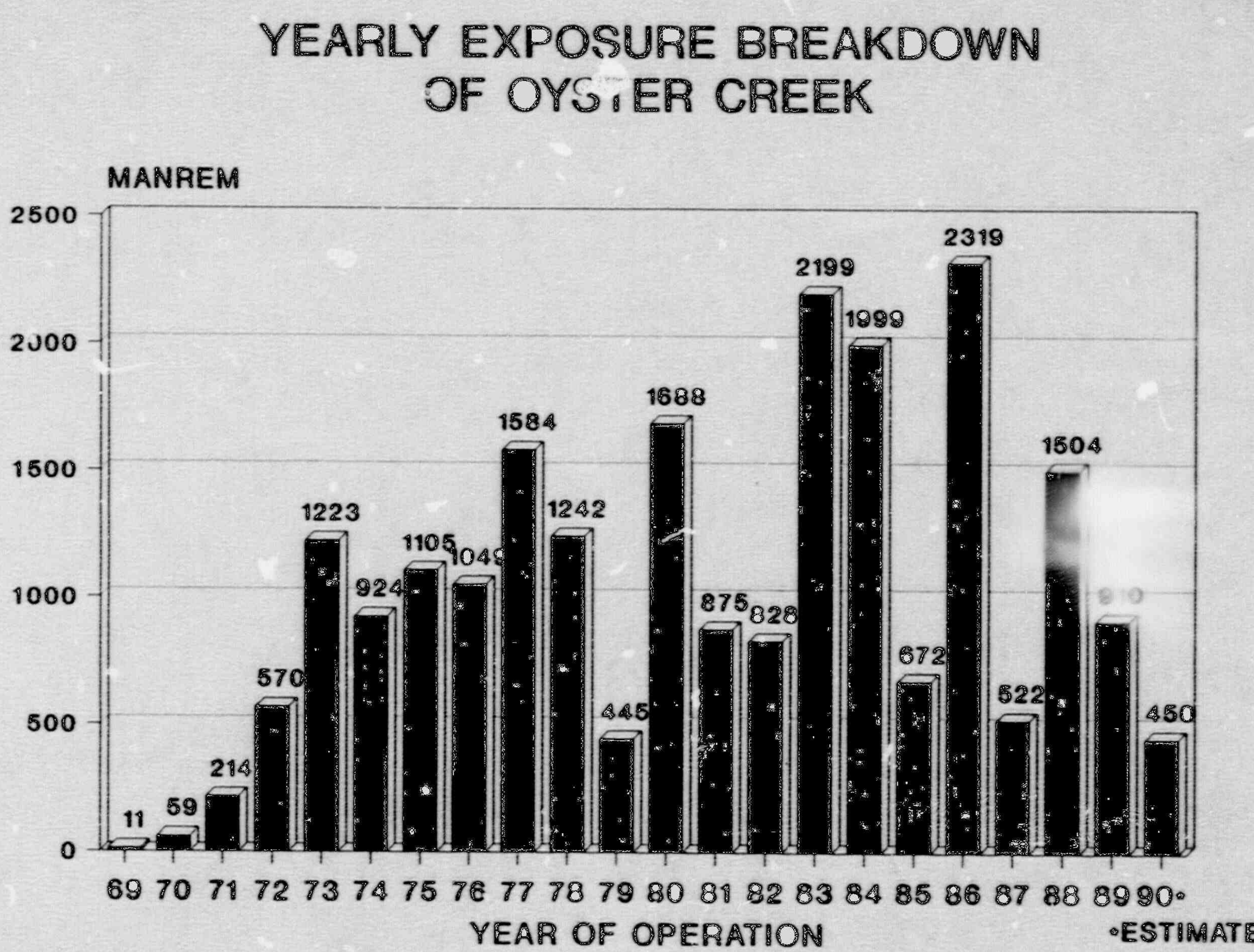
IMPROVEMENT PROGRAM

- **TRAINING & QUALIFICATION**

  - STRENGTHENED GENERAL EMPLOYEE TRAINING
  - NEW PROGRAM FOR ADVANCED RADIATION WORKER TRAINING
  - SUPERVISORS SEMINARS ON RADIOLOGICAL PERFORMANCE
  - RADIOLOGICAL ENGINEERING TECHNICAL TRAINING
  - JOB SPECIFIC MOCK-UP TRAINING INCREASED
  - CRAFT UNION TRAINING PROGRAM UNDER DEVELOPMENT

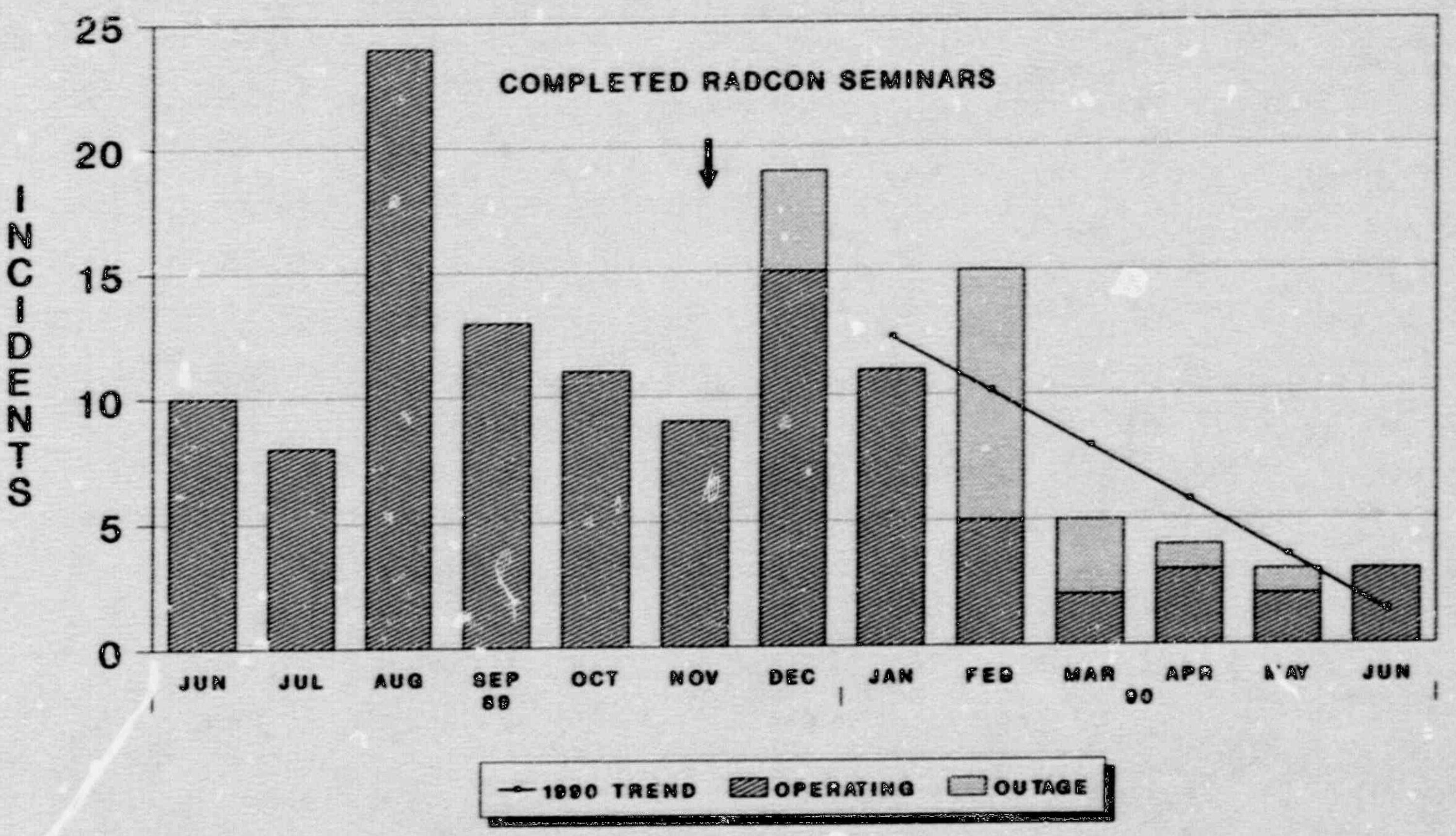
- **OVERSIGHT**

  - QC MONITORING PROGRAM
  - MANAGEMENT TOUR PROGRAM
  - OYSTER CREEK MANAGEMENT TEAM PLAN FOR EXCELLENCE
  - VP LEVEL COLLECTIVE DOSE OVERSIGHT GROUP
  - RADIOLOGICAL & INDUSTRIAL SAFETY ASSESSOR → reports directly to President's office
  - GENERAL OFFICE REVIEW BOARD
  - NUCLEAR SAFETY COMPLIANCE COMMITTEE (REPORTS TO BOARD OF DIRECTORS)

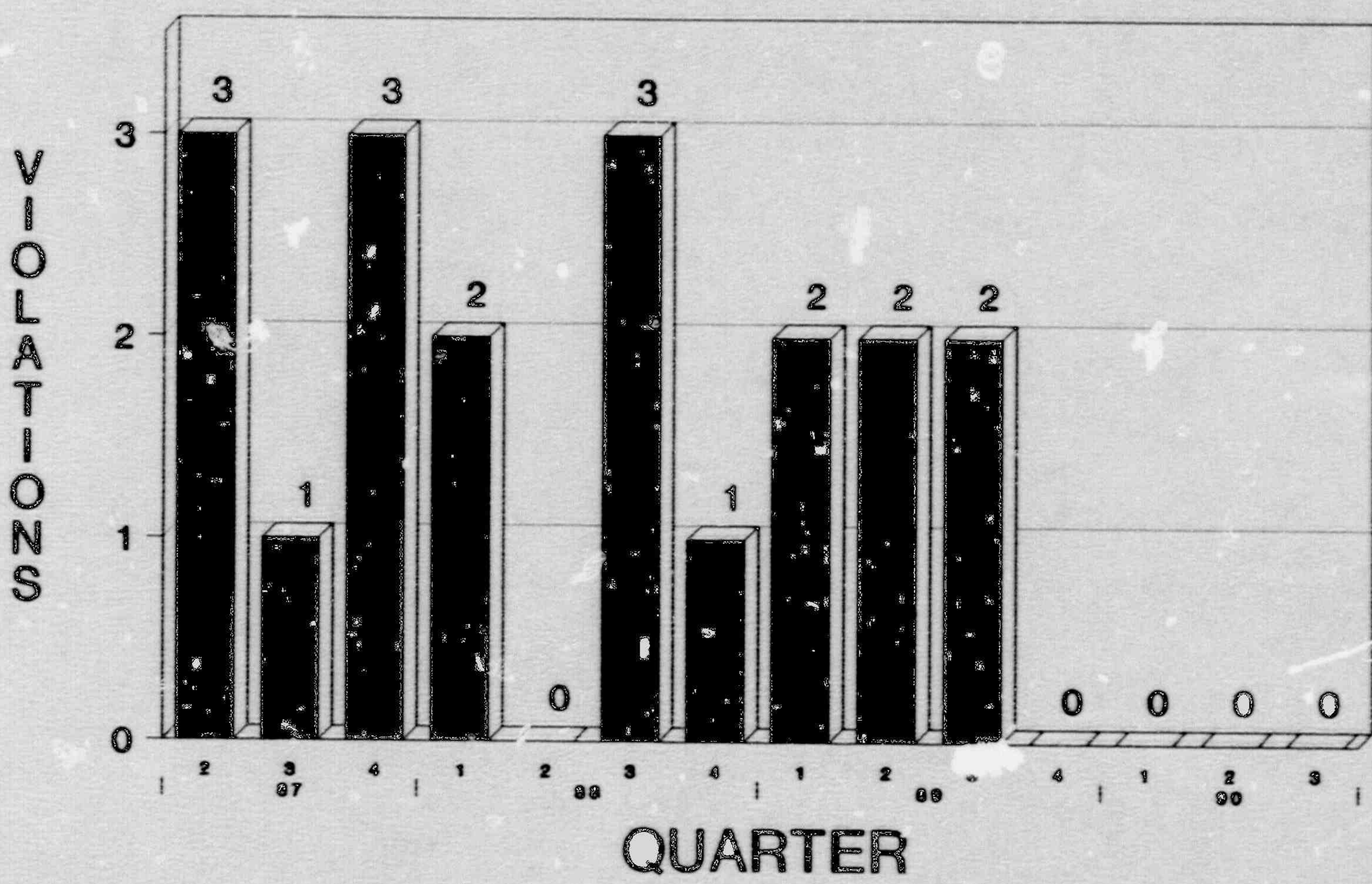


YEARLY EXPOSURE BREAKDOWN
OF OYSTER CREEK
MANREM
2500
2000
1500
1000
500
0
11
59
214
570
1223
924
1105
1049
1584
1242
445
1688
875
828
2199
1999
672
2319
522
1504
910
450
69 70 71 72 73 74 75 76 77 78 79 80 81 82 83 84 85 86 87 88 89 90*
YEAR OF OPERATION
*ESTIMATE

OYSTER CREEK RADCON PARAMETERS: SKIN CONTAMINATIONS

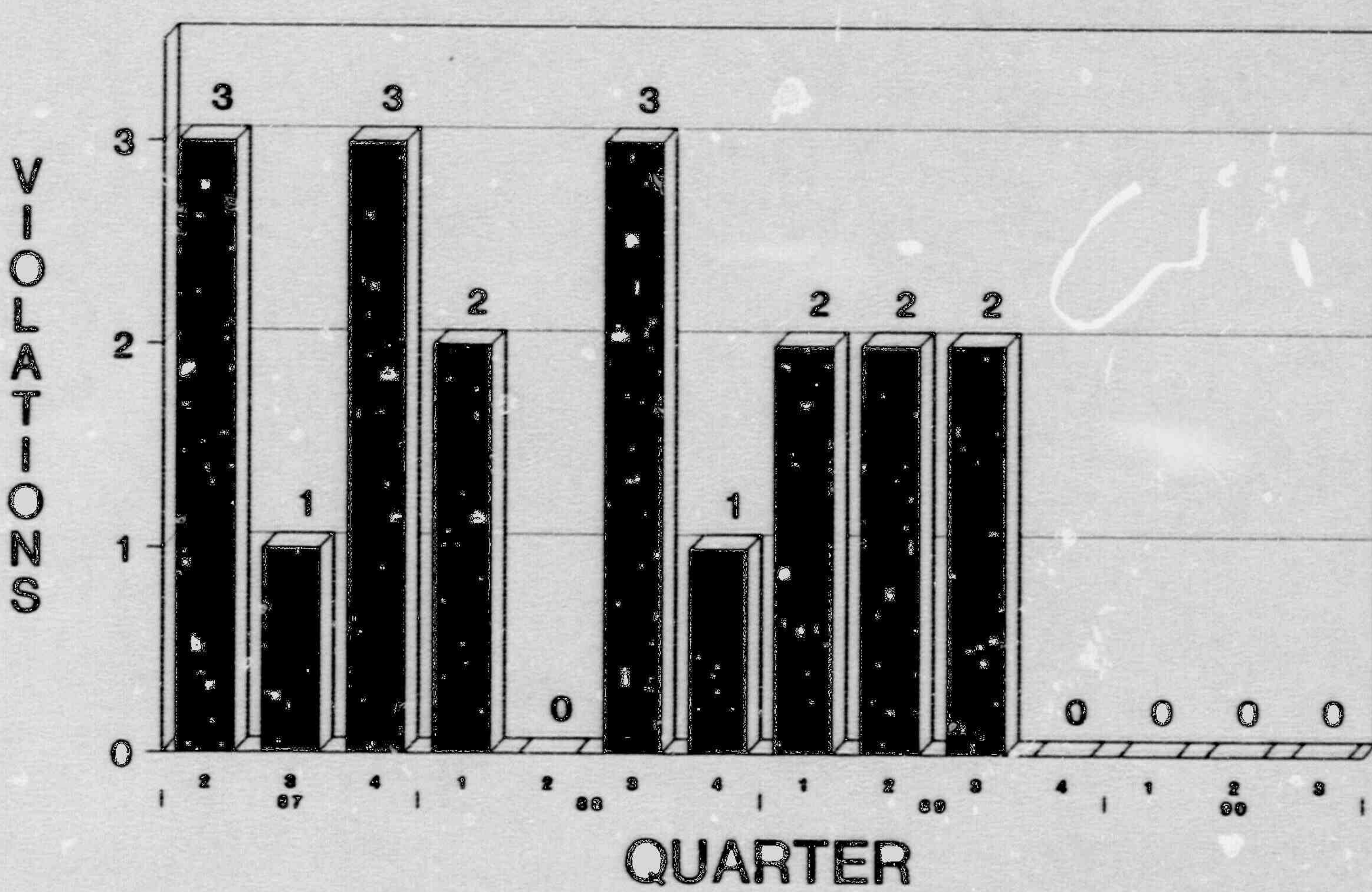


OYSTER CREEK
LOCKED HIGH RAD AREA VIOLATIONS BY QTR

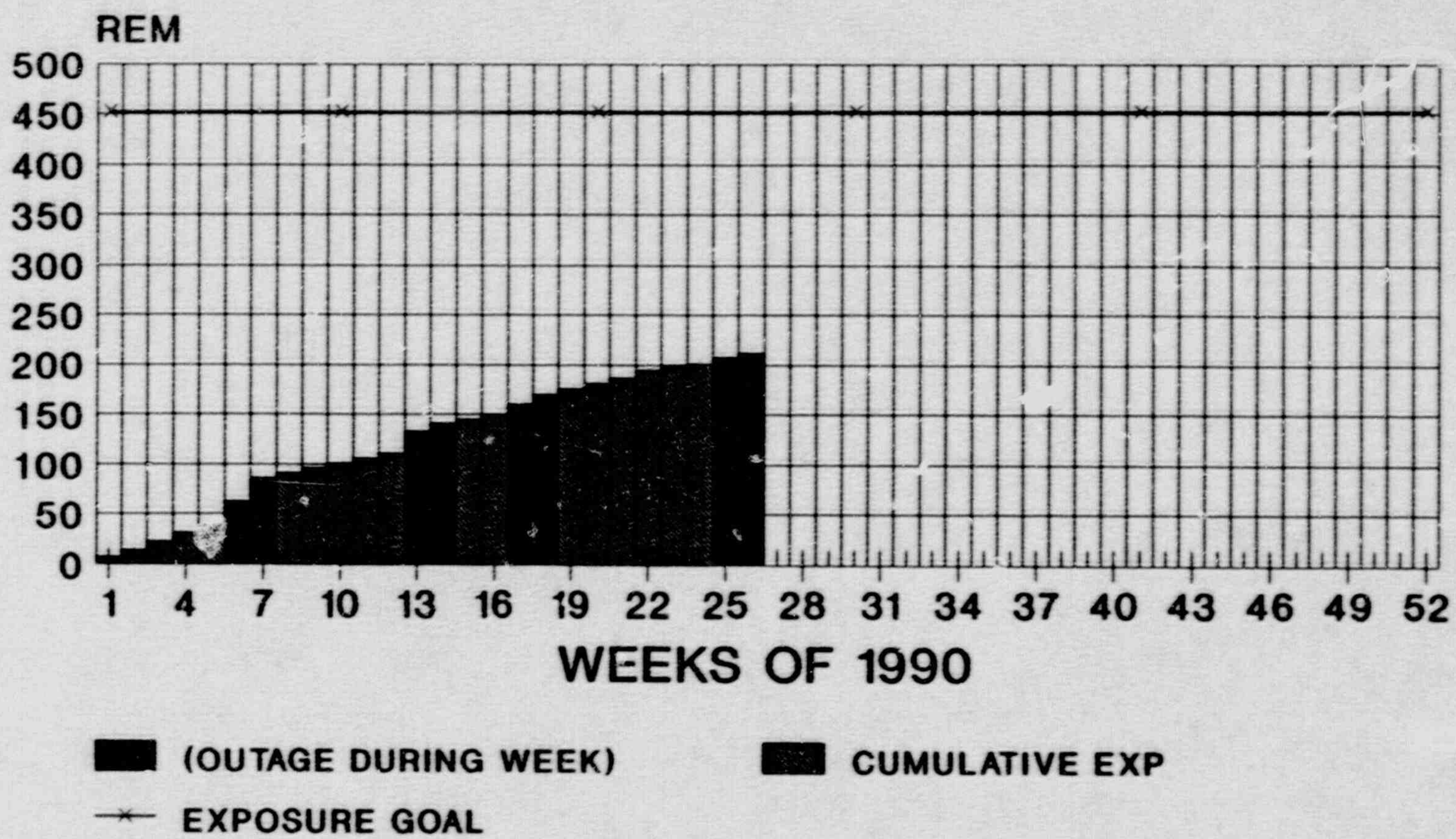


OYSTER CREEK

LOCKED HIGH RAD AREA VIOLATIONS BY QTR



OYSTER CREEK - 1990
Cumulative Exposure Breakdown



MATERIEL CONDITION

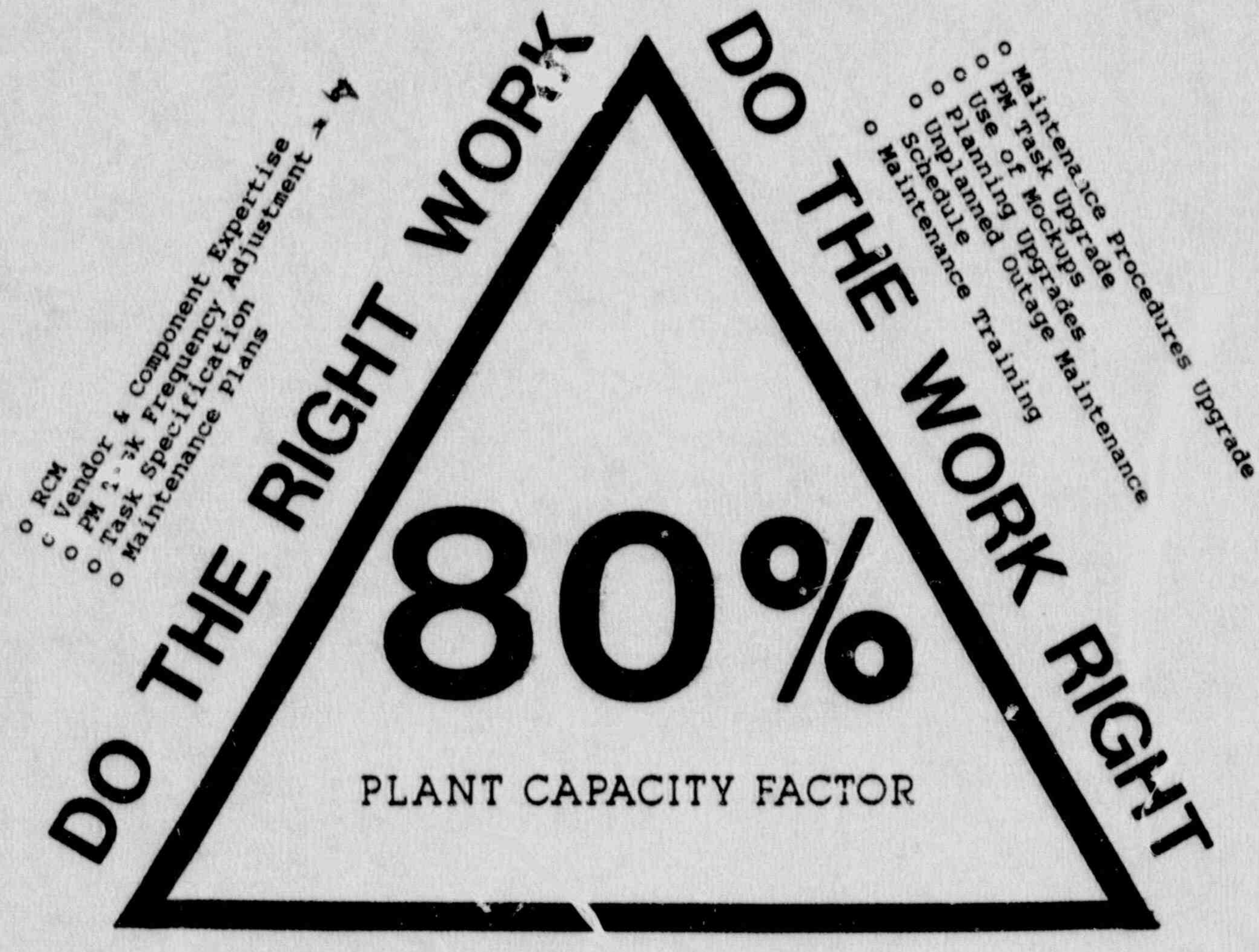
CHALLENGE

CONTINUE TO IMPROVE MATERIEL CONDITION

OF THE PLANT

IN A TIMELY MANNER

LOSMP



FEEDBACK

- o Materiel Condition Issues
- o Failure Trending
- o Condition Monitoring
- o Deviation Reports
- o NPRDS

LIFE OF SYSTEM MAINTENANCE PLAN

IMPROVEMENTS IN RELIABILITY

COMPLETE

- REACTOR RECIRC PUMP/MOTOR/SEAL
- AIR COMPRESSOR
- FEEDWATER PUMP MOTOR
- INTAKE SCREENS
- CONDENSATE PUMP AND MOTOR
- CIRCULATING WATER PUMP
- VALVES
- LPRM POWER SUPPLIES
- SNUBBER SEAL LIFE EXTENSION
- ISO CONDENSER VALVES
- RADWASTE EVAPORATOR TO DEMIN
- AUGMENTED OFF-GAS BLOWERS

LIFE OF SYSTEM MAINTENANCE PLAN

IMPROVEMENTS IN RELIABILITY

<u>**IN-PROGRESS**</u>

- REFUELING BRIDGE
- CONTROL AIR DRYERS
- REACTOR PROTECTION SYSTEM SWITCH
- HFA RELAYS
- CR 120 RELAYS
- AIR OPERATED VALVE DIAPHRAGM
- CRD HCU MODULES

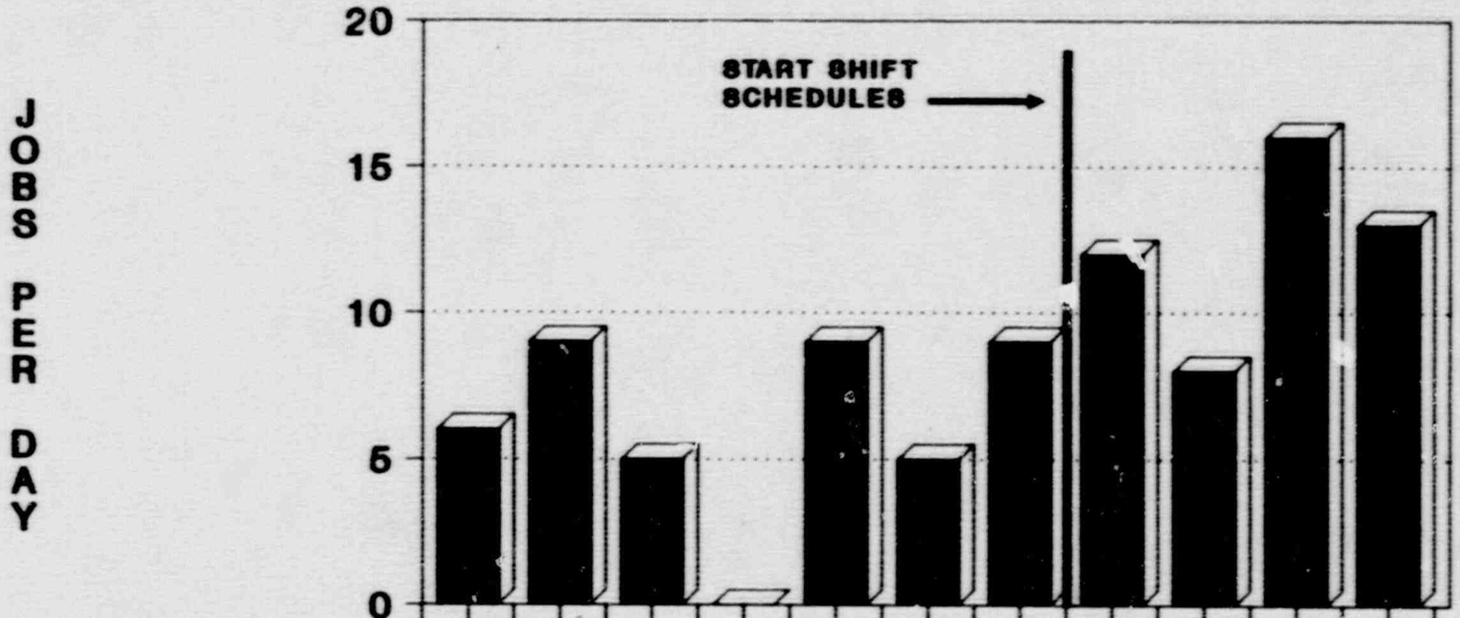
MATERIEL CONDITION

RECENT TRENDS

° RECENT TRENDS (1990)

- WORKFORCE PRODUCTIVITY (IMPROVING)
- MAINTENANCE BACKLOG (IMPROVING)
- MAINTENANCE BACKLOG OLDER THAN 3 MONTHS (IMPROVING)
- RATIO PREVENTIVE/TOTAL MAINTENANCE (IMPROVING)
- CONTAMINATED AREA (LOWEST IN 10 YEARS)
- HOUSEKEEPING IMPROVEMENTS
- PAINTING (60,000 SQ. FT.)

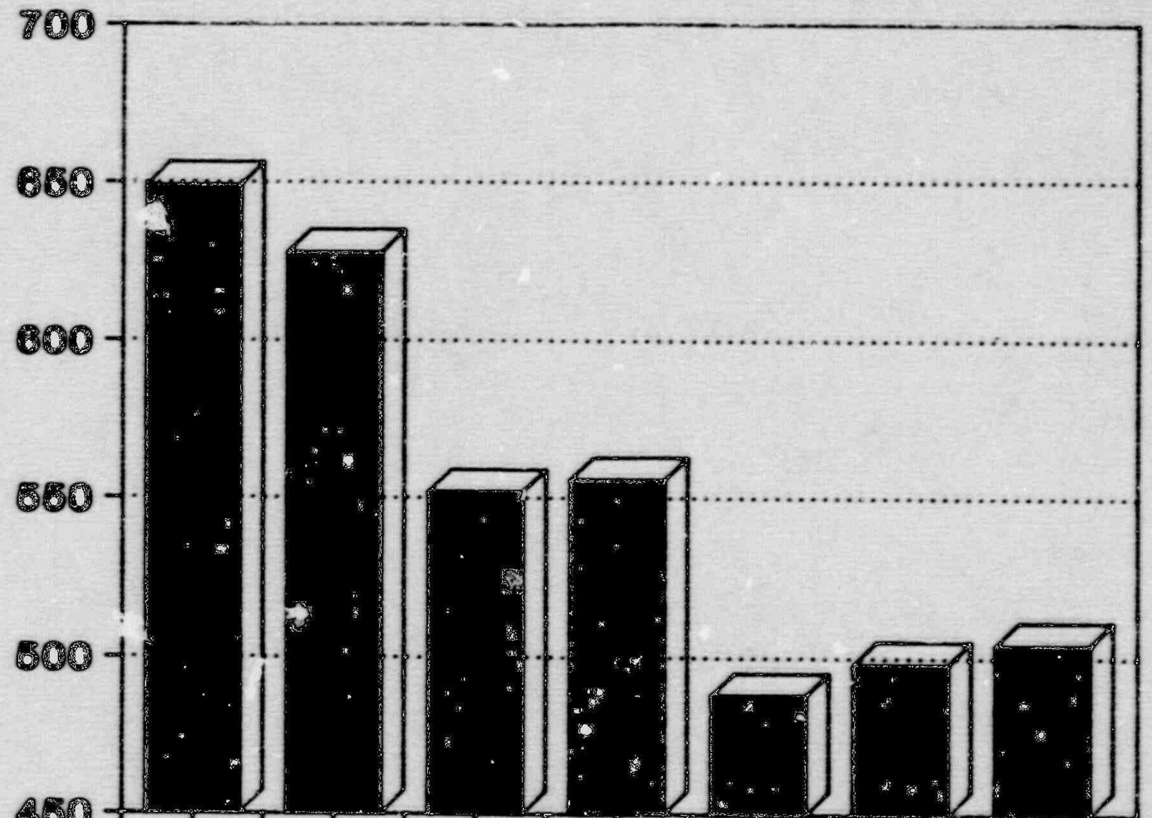
WORK FORCE PRODUCTIVITY
FUNCTIONAL MAINTENANCE COMPLETED DURING CYCLE 12 FORCED OUTAGES



	12U 1	12U 2	12U 3	12U 4•	12U 5	12U 6	12U 7	12U 8	12U [illegible]	12U J	12U K
JOBS/DAY	6	9	5	0	9	5	9	12	8	16	13
DAYS/OUTAGE	5	6	9	1	10	4	9	11	3	9	13

• NO MAINTENANCE WAS APPROVED FOR 12U-4

TOTAL MAINTENANCE BACKLOG (cm)

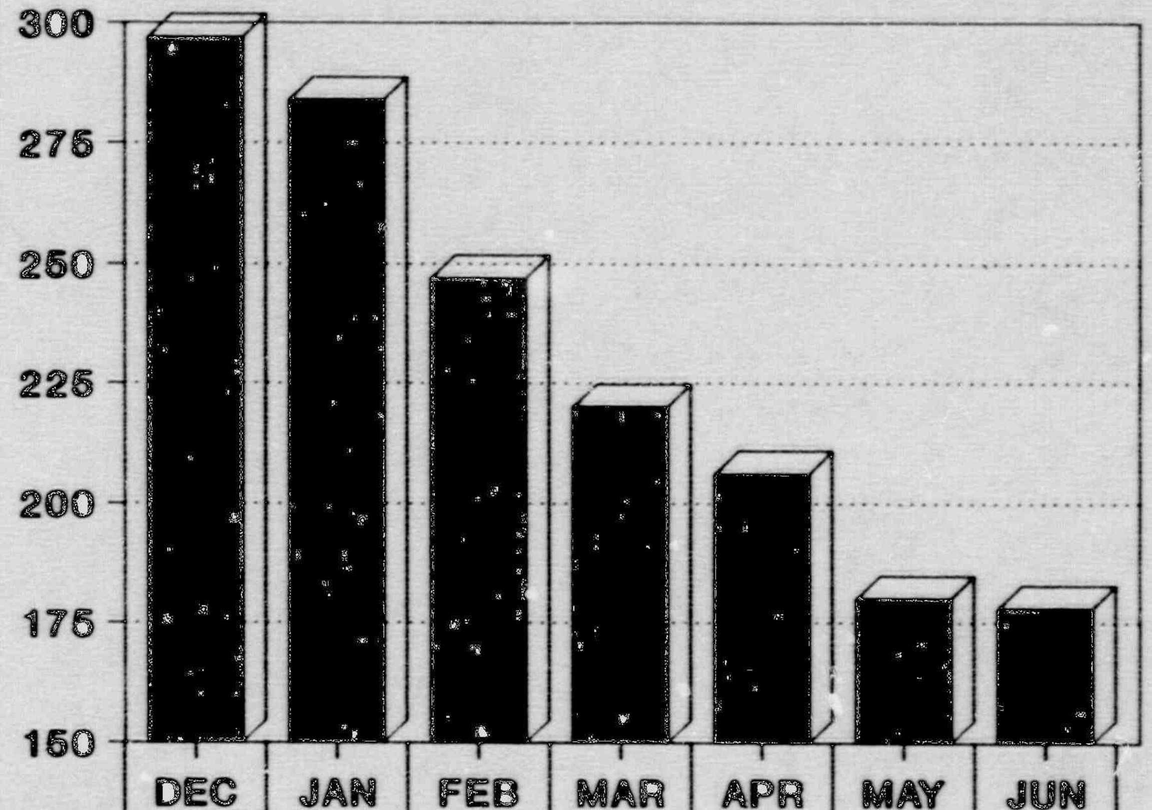


	DEC	JAN	FEB	MAR	APR	MAY	JUN
JOBS ORDERS	649	628	552	556	488	498	504

1989 - 1990

NON OUTAGE JOBS ONLY

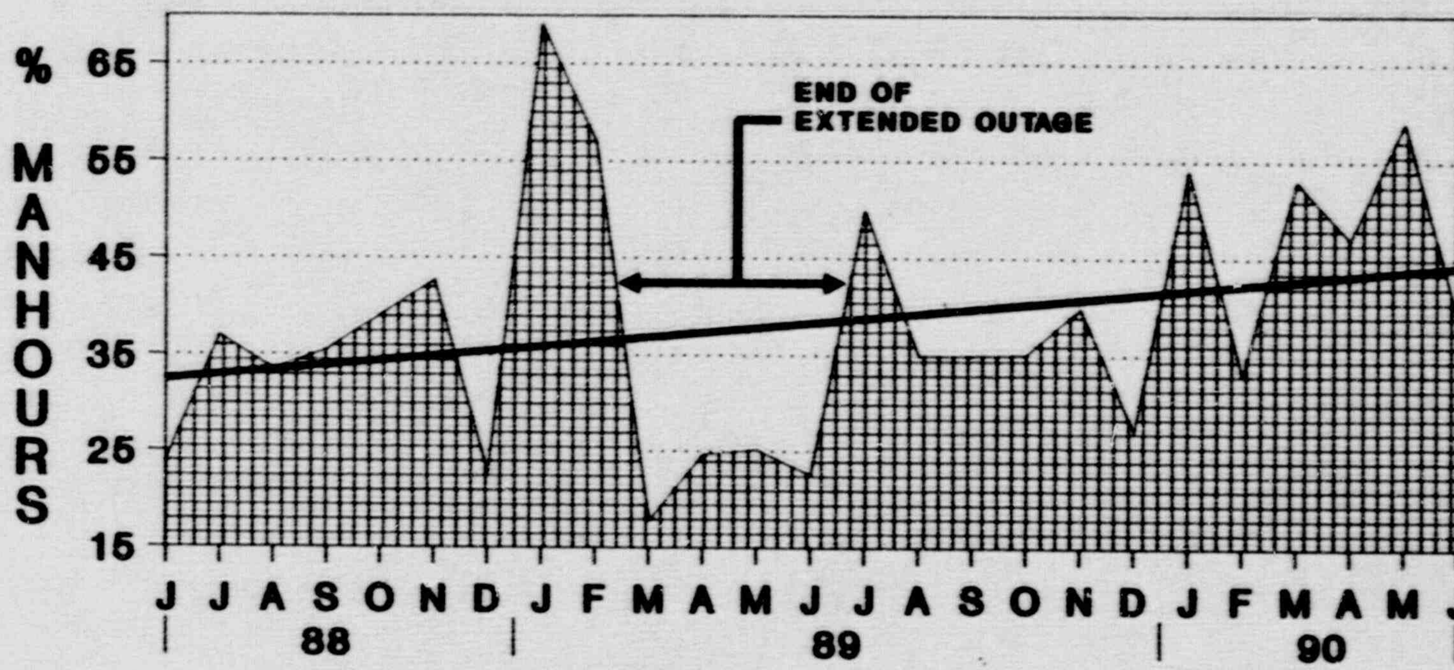
MAINTENANCE BACKLOG
(OLDER THAN 3 MONTHS)



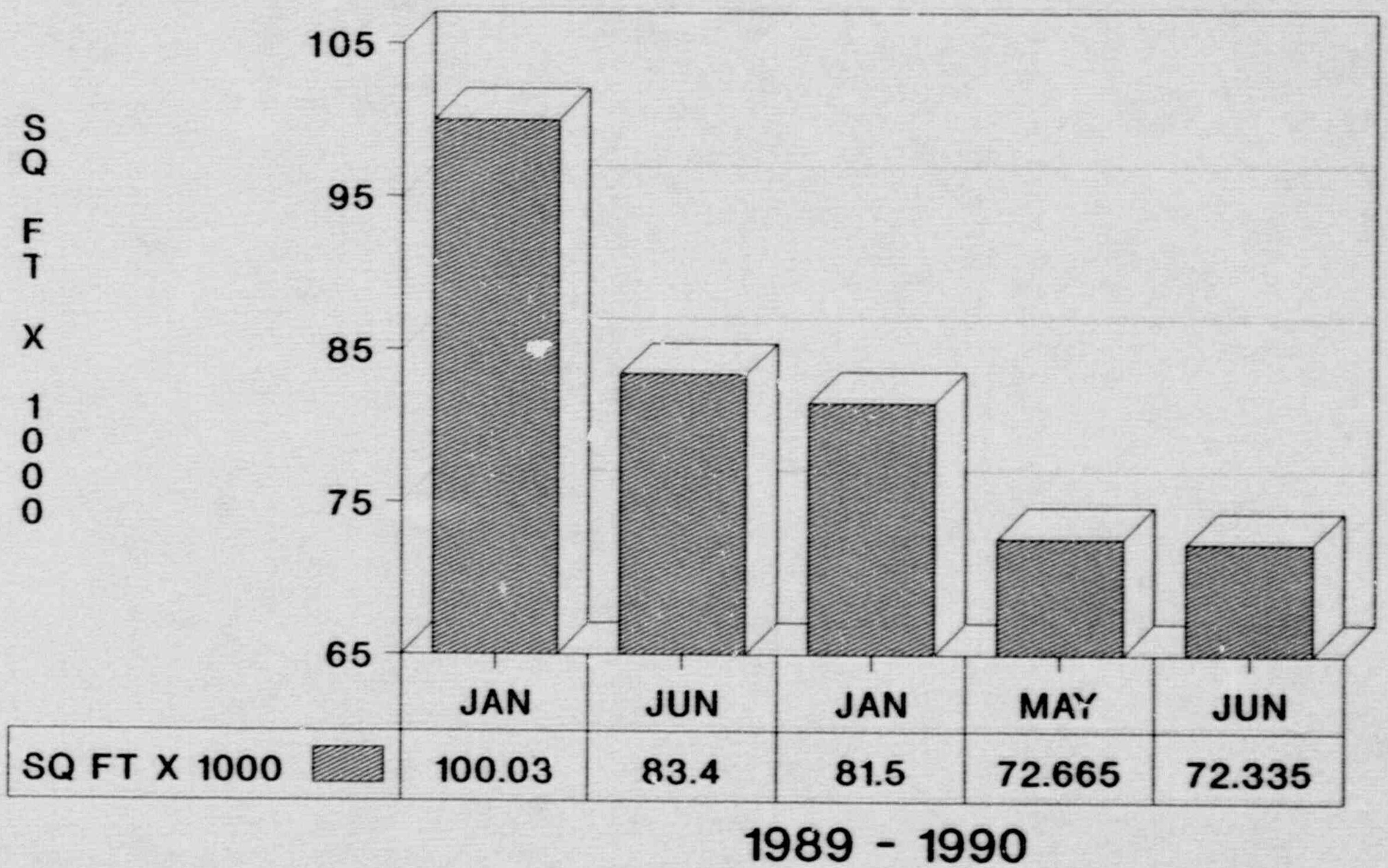
	DEC	JAN	FEB	MAR	APR	MAY	JUN
JOBS ORDERS	297	284	247	220	206	180	178

1989 - 1990

RATIO OF COMPLETED PREVENTIVE TO TOTAL MAINTENANCE



TOTAL CONTAMINATED AREA
SQUARE FEET



CONCLUSION

GPUN IS COMMITTED TO APPLY THE REQUIRED RESOURCES

TO CONTINUE TO IMPROVE AND MAINTAIN THE

THE MATERIEL CONDITION OF OYSTER CREEK