U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-213/90-20

License No. DPR-61

Licensee: Connecticut Yankee Atomic Power Company
P. O. Box 270
Hartford, CT 06141-0270

Facility: Haddam Neck Plant

Location: !addam Neck, Connecticut

Inspection Dates: December 19, 1990 to January 29, 1991

Reporting Inspector: Andra A. Asars, Resident Inspector

Inspectors: Andra A. Asars, Resident Inspector
John T. Shedlosky, Senior Resident Inspector
Alfred Lohmeier, Reactor Engineer, DRS

Approved by: Donald R. Haverkamp 2/7/91
Donald R. Haverkamp, Chief Date
Reactor Projects Section 4A
Division of Reactor Projects

Inspection Summary: Inspection on December 19, 1990 to January 29, 1991 (Inspection Report No. 50-213/90-20)

Areas Inspected: Routine safety inspection by the resident inspectors. Areas reviewed included plant operations, power reductions due to service water filter blockage and apparent malfunctioning of the main steam line trip valves, a pressurizer pressure transmitter failure, radiological controls, auxiliary feedwater pump corrective maintenance, surveillance testing, security, design changes to the main steam line trip valve actuation devices, refueling water storage tank leakage, plant operation review committee meetings, written reports, and licensee self-assessment.

Results: See Executive Summary

EXECUTIVE SUMMARY

Haddam Neck Plant

NRC Region I Inspection No. 50-213/90-20

Plant Operations

Operator response to plant events was evaluated as good. Prompt actions were taken in response to blockage of the service water filters and failure of a pressurizer pressure channel. Operators conservatively declared the main steam line trip valves inoperable when there was an apparent malfunction during surveillance testing. Investigation revealed that no malfunction occurred but the test procedure deficiency was identified and corrected.

Radiological Controls

Good radiological controls performance was noted during this inspection period.

Maintenance and Surveillance

Maintenance responded promptly to the two auxiliary feedwater pump failures. Safety tagging, contractor control, and procedure adherence were good and management presence and support were noted.

Security and Safeguards

Good security performance was noted during this inspection period.

Engineering and Technical Support

The licensee actions taken in response to the refueling water storage tank leakage were found to be acceptable. A program has been implemented for characterization of the leak rate and contingency measures should the leak worsen. Good corporate support was evident in reviewing the tank structural integrity and safety significance of the leakage.

Safety Assessment and Quality Verification

Two licensee-identified, non-cited violations were reviewed regarding the failure to establish a fire watch for an inoperable fire door (50-213/90-20-01) and exceeding the surveillance frequency for fire protection seals (50-213/90-20-02).

A licensee-requested management meeting was held on January 22, 1991 to discuss the licensee programs for self-assessment and various plant initiatives for this operating cycle and upcoming outages. This meeting followed detailed presentations of those matters on

Executive Summary

December 19, 1990 during onsite discussions by corporate and plant managers, department heads and support staff with the resident inspectors, NRR project manager and Region I section chief. Licensee presentations were thorough and informative.

TABLE OF CONTENTS

ATTACHMENTS

DETAILS

1. SUMMARY OF FACILITY ACTIVITIES

Power operations continued through this inspection period. Two technical specification (TS) required power reductions were made in response to equipment problems.

On December 27, 1990, the two service water filters became blocked by sediment which resulted in the inoperability of the containment air recirculation units. A plant shutdown was initiated in accordance with TS 3.0.3. The power reduction was stopped when one filter was returned to service; reactor power was about 73%. The second filter was cleaned and returned to service; full power operations resumed on December 28.

A second TS required plant shutdown was initiated on January 11, 1991. The shutdown was initiated when the four main steam line trip valves (MSTVs) did not operate as expected during a routine surveillance test. The test method was corrected and the MSTVs tested satisfactorily. The plant power reduction was discontinued at about 96% power and 100% power operation resumed the same day.

A meeting was held between licensee management and NRC Region I and NRR managers on January 22, 1991. This meeting followed onsite presentations and discussions on December 19, 1990 by corporate and plant managers, department heads and support staff. This is discussed in detail 8.1 of this report.

In addition to normal utility working hours, the review of plant operations was routinely conducted during portions of backshifts (evening shifts) and deep backshifts (weekend and night shifts). Inspection coverage was provided for 14 hours during backshifts and 8 hours during deep backshifts.

2. PLANT OPERATIONS (71707, 71710 and 93702)

2.1 Operational Safety Verification

The inspectors observed plant operation and verified that the plant was operated safely and in accordance with licensee procedures and regulatory requirements. Regular tours were conducted of the following plant areas:

-- control room	-- security access point
-- primary auxiliary building	-- protected area fence
-- radiological control point	-- intake structure
-- electrical switchgear rooms	-- diesel generator rooms
-- auxiliary feedwater pump room	-- turbine building

Plant areas were observed to be in generally good condition.

2.2 FOLLOW-UP OF EVENTS OCCURRING DURING INSPECTION PERIOD

During the inspection period, the inspectors provided on-site coverage and follow-up of unplanned events. Plant conditions, alignment of safety systems, and licensee actions were reviewed. The inspectors confirmed that required notifications were made to the NRC. During event follow-up, the inspectors reviewed the corresponding plant information report (PIR) package, including the event details, root cause analysis, and corrective actions taken to prevent recurrence.

2.2.1 POWER REDUCTION DUE TO SERVICE WATER FILTER BLOCKAGE

On December 26, the plant began experiencing increased service water filter blockage. These filters supply cooling water to the containment air recirculation (CAR) heat exchangers. Full power operations continued and the two filters were alternately cleaned by the installed backwash system. This system of cleaning was adequate until December 27 at 1:35 p.m. when the inservice filter also became blocked. This rendered all four of the CAR units inoperable.

Technical Specification (TS) 3.6.2 provides the requirements for the CAR units but does not address compensatory measures when all four CAR units are out of service. Therefore, the licensee entered TS 3.0.3 and initiated a plant shutdown at 1:35 p.m. on December 27.

One filter was tagged out for mechanical cleaning by maintenance personnel; the other filter continued to backwash. At 2:45 p.m. maintenance personnel returned one filter to service providing filtered water to the CAR units. The load reduction was terminated at about 73% power and TS 3.0.3 was exited.

The second filter was tagged out for mechanical cleaning several hours later, cleaned and promptly returned to service. Both filters were fully operable and back in service by 11:11 p.m. on December 27.

Full power operations were resumed on December 28.

2.2.2 POWER REDUCTION DUE TO APPARENT MALFUNCTION OF MAIN STEAM LINE TRIP VALVES

On January 11, during the routine surveillance testing of the main steam line trip valves (MSTVs), the valves failed to operate as expected. The valves were declared inoperable at 12:30 p.m. and a plant shutdown was initiated in accordance with technical specification (TS) 3.7.1.5. The appropriate notifications were made and troubleshooting activities started.

SUR 5.1-12, "Main Steam Line Isolation Trip Valve Test", is a quarterly partial stroke test of the MSTVs to verify that they will begin to close upon receipt of a test signal. The test switch operates a normally closed solenoid-operated valve (SOV) to provide closing air pressure to the valve operator. During this performance of SUR 5.1-12 when the SOV were operated the associated MSTVs did not move in the closed direction as expected. The valves were conservatively declared inoperable during investigation of this event.

The licensee determined that the test switch had not been held long enough for sufficient air pressure to build into the valve operating system and initiate valve movement. Extended test switch depression is a new requirement following a modification implemented during the recent refueling outage. This is described in section 6.1 of this report.

The valves retested satisfactorily when the test switch was held for about 15 seconds. All four valves were declared operable at 1:38 p.m. and a load increase to full power was initiated. Temporary Procedure Change No. 91-0015 was issued to add a note specifying the test switch operation time.

2.2.3 PRESSURIZER PRESSURE TRANSMITTER FAILURE

On January 21, at 10:26 a.m. channel No. 2 of pressurizer pressure failed high. The channel was declared inoperable and the required actions of technical specifications (TS) were initiated. TS Tables 3.3-1 and 3.3-2 permit continued power operation provided that the affected channel be placed in the trip condition. Trip signals were inserted for the high pressurizer pressure and variable low pressurizer pressure reactor trips and the low pressurizer pressure safety injection initiation at 11.22 a.m. A trip signal was also inserted for this channel's input into the pressurizer power-operated relief valve actuation circuitry. The more restrictive requirements of the TS action statements requirements of the TS action statements require the channel to be placed in the trip condition within one hour (TS 3.3-2) and permit power operation until the next required analog channel operational test (TS 3.3-1). The analog operational test frequency is once per 42 days and the last test was performed on December 31, 1990.

The affected channel's amplifier was replaced and channel performance was monitored until January 23. The channel had not exhibited any noticeable fluctuations and the trip signals were removed at 11:04 a.m.

Power operations continued without incident until January 28. At 1:57 p.m. pressurizer pressure channel No. 2 was observed to drift low. The channel was again declared inoperable and the required trip signals were inserted at 2:16 p.m.

Review of the control room instrument chart for this channel for the eight day period of operation displayed a slow downward drift. The licensee determined that the transmitter had failed. At the end of the inspection period, the trip signals remained inserted and the time response testing of the spare transmitter was ongoing.

The inspectors reviewed the licensee's responses to the channel failures and verified that the requirements of TS were met for insertion of trip signals and channel surveillance frequencies.

2.2.4 EMERGENCY OPERATING PROCEDURES REGARDING REACTOR COOLANT PUMP SEAL COOLING

Following the 1979 accident at the Three Mile Island (TMI) Nuclear Plant, the Commission generically reviewed the potential for failure of reactor coolant pump (RCP) seals following a loss of off-site power. This led to the establishment of the proposed TMI Task Action Plan (TMI-TAP) item II.K.3.25 (in NUREG 0737). This item requires the licensees to evaluate the integrity of the RCP seals for a two-hour period following a loss of off-site power.

The initial resolution of this item at Haddam Neck was based on operator actions to restore seal cooling. However, this position was not accepted by the NRC staff until a commitment was made to automate the reinitiation of RCP seal water flow following restoration of off-site power (see letters: Connecticut Yankee Atomic Power Company (CYAPCO) to NRC dated December 31, 1983, April 5, 1983, October 4, 1984, May 17, 1985, and June [illegible], 1985 and NRC to CYAPCO dated October 31, 1980, June 7, 1982, February 1, 198[illegible] and May 1, 1985).

CYAPCO's position was summarized in a letter dated December 5, 1990, addressing the results of their evaluation of RCP seal integrity. This was based on pump and seal design, the design basis for seal cooling, and emergency procedures for restoration of seal cooling. Additionally, test results and the results of risk assessment studies were addressed. CYACPCO concluded that additional actions were not required.

The NRC staff has not concluded its review of the licensee's position. However, currently CYAPCO credits the effectiveness of certain emergency procedure actions to restore RCP seal cooling. For this reason, the inspector reviewed the emergency procedures as they relate to this issue. The appropriate procedures changes have been made. But, the step numbers identified in the December 5, 1990 letter are different from those in EOP 3.1-10, "Partial Loss of AC." This has no safety impact.

The inspector reviewed the following procedures: EOP 3.1-10, Partial Loss of AC, Revision 14, dated May 11, 1990; ES-0.1, Reactor Trip Response, Revision 5, dated April 9, 1990 [illegible]th Temporary Procedure Change No. 90-668, dated July 14, 1990; ECA-0.0, Station [illegible]t, Revision 6, April 9, 1990; ECA-0.1, Station Blackout Recovery Without SI [illegible]ed, Revision 4, dated April 9, 1990; and, ECA-0.2, Station Blackout Recovery with SI [illegible]ired, Revision 6, dated April 9, 1990.

[illegible]itional evaluation of site procedures or equipment may be required following final NRC staff review of the licensee's position.

3. RADIOLOGICAL CONTROLS (71707)

During routine inspections of the accessible plant areas, the inspectors observed the implementation of selected portions of the licensee's radiological controls program. Utilization and compliance with radiation work permits (RWPs) were reviewed to ensure that detailed descriptions of radiological conditions were provided and that personnel adhered to RWP requirements. The inspectors observed controls of access to various radiologically controlled areas and the use of personnel monitors and frisking methods upon exit from those areas. Posting and control of radiation areas, contaminated areas and hot spots, and labelling and control of containers holding radioactive materials were verified to be in accordance with licensee procedures. During this inspection period health physics technician control and monitoring of station activities were determined to be good.

4. MAINTENANCE AND SURVEILLANCE (61726, 62703 AND 71707)

4.1 MAINTENANCE OBSERVATION

The inspectors observed various corrective and preventive maintenance activities for compliance with procedures, plant technical specifications, and applicable codes and standards. The inspectors also verified the appropriate quality services division (QSD) involvement, use of safety tags, equipment alignment and use of jumpers, radiological and fire prevention controls, personnel qualifications, post-maintenance testing, and reportability. Portions of activities that were reviewed included:

-- Troubleshooting and repair of "B" auxiliary feedwater pump

-- Fabrication of a new containment air recirculation damper linkage

Maintenance activities were determined to be acceptable; the following observations were made.

4.1.1 AUXILIARY FEEDWATER PUMP CORRECTIVE MAINTENANCE

During this inspection period there were two failures of the "B" auxiliary feedwater (AFW) pump during surveillance testing. In both cases the turbine tripped on overspeed, however, the licensee has concluded that the events were not related.

On January 2, at 8:50 a.m., in preparation for maintenance on the "A" train of AFW, the "B" AFW pump was tested to verify operability. The test performed was the routine monthly surveillance test, SUR 5.1-13B, "Auxiliary Feed Pump (P-32-1B) Monthly Functional Test." As operators were increasing turbine speed by opening the steam admission valve, the turbine overspeed mechanism actuated with steam pressure at about 420 psig. The pump was declared inoperable, technical specification (TS) 3.7.1.2 was entered and the required action was taken in accordance with Operations Department Instruction 169, "Administrative Controls for Plant Operations." For this situation, a dedicated auxiliary operator is provided should it become necessary to operate the electric auxiliary feedwater pump.

Maintenance and engineering personnel immediately initiated troubleshooting activities. The pump was run two additional times to observe system parameters prior to the turbine overspeed trip. The turbine tripped during both test runs at about 420 psig steam inlet pressure. A work order was initiated to inspect the overspeed trip mechanism and make any necessary adjustments. Maintenance personnel observed that the lock nut for the overspeed adjustment screw had become loose. It was determined that this caused the apparent lowering of the overspeed trip setpoint. The lock nut was tightened and the overspeed trip setpoint verified prior to returning the pump to service at 1:15 a.m. on January 3.

The second "B" AFW pump failure occurred on January 15, at 9:13 a.m. during the routine monthly performance of SUR 5.1-13B. In this case, the turbine steam admission pressure was normally increased to 450 psig without incident. However, as operators were decreasing steam pressure, the overspeed trip mechanism actuated at about 400 psig steam pressure. The pump was declared inoperable, the appropriate administrative actions were taken, and troubleshooting initiated.

Maintenance and engineering personnel, supplemented by the turbine vendor representative, observed system parameters and overspeed mechanism actuation during two subsequent pump tests. In both tests, the overspeed trip device actuated when steam admission pressure reached about 400 psig as the pump was being slowly started.

The overspeed trip mechanism is a spring loaded trip lever which actuates the turbine governor valve on an overspeed condition. The vendor representative observed that the trip lever made contact with the emergency trip rod in a higher position than normal. It was also observed that during the two test runs the emergency trip rod appeared to vibrate during pump operation and cause the overspeed trip device to actuate. It was noted that these surfaces are painted and that where contact is made the paint had worn away creating a groove. In accordance with the vendor recommendation, the licensee removed the paint from the contact surfaces of the trip mechanism and retested the pump.

The "B" AFW pump was started and run in accordance with SUR 5.1-13B for several minutes with a steam admission pressure of 450 psig. No overspeed trip occurred and the overspeed trip lever was observed to be correctly positioned and to not vibrate as in the previous tests. The pump was shutdown and an overspeed trip setpoint check was performed to verify that the setpoint had not been affected. The test was successful; the setpoint had not been altered. SUR 5.1 13B was performed as the post maintenance test and the pump was returned to service on January 15, at 10:45 p.m.

The inspector observed troubleshooting, repair, and retest activities following the failure on January 15. Safety tagging, contractor control, and procedure adherence were observed to be good. The inspector noted management presence and support during this evolution.

The overspeed adjustment lock nut on the "A" AFW pump was inspected and verified to be secure. At the end of the inspection period, the licensee was planning maintenance activities for the "A" AFW pump to clean and inspect the overspeed trip mechanism to preclude similar premature tripping due to vibration and worn paint surfaces.

4.2 SURVEILLANCE OBSERVATION

The inspectors witnessed selected surveillance tests to determine whether: properly approved procedures were in use; plant technical specifications frequency and action statement requirements were satisfied; necessary equipment tagging was performed; test instrumentation was in calibration and properly used; testing was performed by qualified personnel; and, test results satisfied acceptance criteria or were properly dispositioned. Portions of activities associated with the following procedures were reviewed:

-- SUR 5.1-17A, "Emergency Diesel Generator EG-2A Manual Starting and Loading Test"

-- SUR 5.1-13B, "Auxiliary Feed Pump (P-32-1B) Monthly Functional Test"

Surveillance activities were determined to be acceptable.

5. SECURITY (71707)

During routine inspection tours, the inspectors observed implementation of portions of the security plan. Areas observed included access point search equipment operation, condition of physical barriers, site access control, security force staffing, and response to system alarms and degraded conditions. These areas of program implementation were determined to be good.

6. ENGINEERING AND TECHNICAL SUPPORT (37700, 37828 and 71707)

The inspectors reviewed selected engineering activities. Particular attention was given to safety evaluations, plant operations review committee approval of modifications, procedural controls, post-modification testing, procedures, operator training, and Updated Final Safety Analysis Report and drawing revisions.

6.1 MAIN STEAM LINE TRIP VALVE OPERATOR MODIFICATION

The MSTVs are 24-inch, reverse-seated check valves with a valve stem connected at the disc center to an operating cylinder. Normally, control air is supplied under the operating piston to keep the valve open against spring pressure. When a closed signal is generated, a solenoid-operated valve (SOV) vents the control air from under the piston and control air pressure is supplied to the top of the piston to assist the spring in shutting the valve. During a MSTV partial stroke test the control air is provided to the top of the piston but is not vented from under the piston (a rod is placed on the operator stem to prevent full valve closure).

During the recent refueling outage, the licensee implemented a modification to the main steam line trip valve (MSTV) operators. Plant Design Change Record Evaluation (PDCE) 88-76, "Replace MSTV Solenoid Valves", provided for replacement of eight SOVs and portions of the associated air tubing. Several of the SOVs were replaced with valves of a different design and seating materials and some of the associated copper tubing was replaced with stainless steel tubing. The post modification test plan included full and partial stroke tests of the MSTVs to verify system operability. This testing was successfully performed during the refueling outage.

At that time, operations personnel observed that during the partial stroke testing the time elapsed between actuating the test switch and valve movement was slightly longer than previously experienced. This was attributed to the lack of system steam since the plant was shutdown. Engineering personnel did not anticipate that the PDCE would affect the partial stroke test conduct.

The inspector reviewed the PDCE package and noted that although the work was completed and tested and accepted by operations, the PDCE has not been closed out. Items remaining open include update of the Preventive Maintenance Management System and Environmental Equipment Qualification List. This is not an operational concern, however it is prudent to close out administrative items resulting from modifications promptly. The inspectors will continue to monitor modification completions during routine inspections.

6.2 REFUELING WATER STORAGE TANK LEAKAGE EVALUATION

On December 19, a NRC Region I specialist inspector conducted an inspection of the methods utilized by the licensee to evaluate the refueling water storage tank (RWST) leakage and justify tank operability until the next planned maintenance outage in October, 1991, when repairs can be made.

6.2.1 BACKGROUND

The RWST provides for a supply of borated water for emergency core cooling system (ECCS) injection during a loss of coolant accident (LOCA), for emergency boration and for plant makeup during cool down. The RWST is a 230,000 gallon capacity flat-bottomed cylindrical tank with a domed head constructed of welded aluminum plates of thickness varying from 0.250 to 0.504 inch. The cylindrical portion of the tank is 35 feet in diameter. The flat bottom plate is coated with a high quality asphalt paint. It is supported around its periphery by a reinforced concrete foundation with a bed of sand supporting the center region of the plate. Also around the tank periphery are welded 16 seismic support brackets through which pass bolts connecting the RWST to the concrete foundation.

As documented in a plant information report written September 14, 1990, after the unit start-up from the 1989/1990 refueling outage, leakage of borated water was observed from the bottom of the RWST at circumferential locations between the concrete foundation and the tank bottom plate. An approximate leakage rate of six gallons per day was estimated at that time. Since then, the licensee developed an action plan to provide for an evaluation of the leak severity which justified continued operation, compared the leakage problem to a similar case of tank leakage at Vermont Yankee Nuclear Power Station, established a program for trending the leakage rate, provided for a collection device which isolated the leakage and collected it for quantification of the leakage rate, developed a plan for root cause evaluation and evaluated the generic implications of the leakage on similar tanks manufactured by the same vendor.

6.2.2 TANK INSPECTION AND LEAKAGE MEASUREMENT

Since the establishment of the action plan, licensee initiatives included provisions for tank leak collection and trending, structural evaluation of the tank, evaluation of the effect of leakage on the concrete foundation integrity, studies of the radiological contamination effect of the leakage of the borated water, evaluation of the tank seismic anchorage for deterioration, and evaluation of the effect of leakage during a small break LOCA station blackout and Appendix R fire. The Haddam Neck site engineering staff utilized the specialized expertise of the Northeast Utilities Service Company (NUSCO) in implementing elements of the action plan.

The inspector conducted a walkdown with the licensee around the RWST to observe the leakage and method utilized to measure the leakage rate. It was observed that the leakage occurred only in the region of the seismic support brackets, although leakage was not observed at all seismic supports. The largest amount of leakage occurred at two support locations and some leakage occurred at nine of the 16 support locations. There were signs of dark oil-like stains at three separate locations.

The following are the qualitative observations of leakage by the inspector:

Seismic Support	Leak Size	Seismic Support	Leak Size
1	small	9	small
2	oil stain	10	none
3	small	11	small w/oil
4	none	12	none
5	small	13	small
6	none	14	boron residue
7	none	15	heavy
8	none	16	heavy

The leakage was collected in diked areas at the largest leak sites. The leakage collected at the dike locations was pumped at known time intervals into plastic collection bottles. In this manner, the leakage rate was determined. A tent covered the leak collection site to prevent rain from being introduced into the diked area. This collection and measurement method provided reasonable estimates of the leakage rate and was particularly useful in revealing changes in leakage rate.

The leakage rate was measured by the licensee at varying intervals of time since September, 1990, with more frequent measurements being taken as the leakage rate increased. The following leakage rates were recorded by the licensee:

Date	Gallons per Day
September 18	6
October 26	10
November 21	30
December 3	30-35
December 13	50
December 14	50-55
December 18	50
January 25	55

On the basis of these measurements, it is apparent that the leakage is either increasing in magnitude or in numbers of locations.

In addition to monitoring the leakage rate at increased frequencies, the licensee is giving consideration to the possibility of utilizing a submerged ultrasonic detection device to determine the existence of incipient leaks from wastage or cracking corrosion. This will allow inspection without requiring drainage of the tank.

6.2.3 TANK STRUCTURAL INTEGRITY

The licensee discussed the evaluation of the integrity of the RWST by NUSCO specialists. They estimated the flaw size to be 1.5 inches in length on the basis of a leakage rate of seven gallons per day. The licensee stated that the primary stresses at the tank bottom were compressive in nature and that the secondary stresses at the tank-bottom plate edge are significant but self-limiting. The licensee believed that the integrity of the cylindrical portion of the tank did not depend on the integrity of the bottom plate.

The effect of borated water on the foundation was given consideration by the licensee. Inspection of the foundation by the licensee indicated that, although cracks were observed in the foundation, the cracks were tight and typical of normal concrete structures. Originally, there were eight foundation support locations. Later, eight more foundation supports were added as a result of seismic loading reevaluation. The licensee stated that no leakage was observed at the sites of the eight original supports. Leakage was observed at the site of six of the eight added support locations.

In review of the tank drawings, the inspector noted that the bottom plate was constructed by many flat, rectangular plates joined by overlapped plate weld joints which could result in spaces between the bottom plates and sand. Furthermore, discussions with the licensee indicated that during the last inspection inside the empty RWST, it was noted that the plates appeared flexible when walked upon during the bottom plate inspection.

Although the inspector had not reviewed the detailed RWST structural calculations, it appears that evaluation of the relation of the proximity of the leakage areas to those where new seismic supports were added should be suspect as a possible reason for the leakage occurring only at these locations. The licensee did not attribute the leakage location information to any causative factor.

6.2.4 SAFETY SIGNIFICANCE

In evaluating the significance of the leaks and leakage rates, the licensee has considered that, at the present low levels of leakage, the leaks are considered to be only of nuisance effect. The radiological contamination resulting from the leaks was determined to be of little significance (although steps have been taken to prevent their introduction into the external drainage system). The licensee believes that the total loss of all the lightly contaminated borated water from the tank would not be a problem from a radiological point of view.

The licensee has provided for the following safety provisions related to tank leakage consistent with Technical Specification limitations, should a more significant leak rate occur:

(1) The maximum leak rate that can be tolerated without jeopardizing RWST operability is four gallons per minute (5760 gallons per day). At leakage rates in excess of four gallons per minute (gpm) the tank should be declared inoperable and a plant shutdown commenced.

(2) The RWST level must be maintained above 240,000 gallons to ensure sufficient water is available to mitigate a small break LOCA with a four gpm leakage rate. For this reason an inventory of RWST borated water of 240,000 gallons is being maintained.

(3) At a leakage rate of one gpm the operability of the RWST will be reevaluated by the licensee.

(4) The licensee will take leakage rate measurements on a regular basis.

Inspector assessment of the licensee engineering involvement in the RWST leakage evaluation indicated that technical considerations were given to a wide range of issues directed toward assessment of the cause of RWST leakage and justification of operation until the next outage with appropriate safeguards for safe operation if a LOCA should occur.

The site engineering staff utilized the expertise of the NUSCO engineering staff to evaluate many of the technical issues. The inspector was assured that the expertise would be called upon should the site staff deem it necessary.

6.2.5 CONCLUSION

Based on this inspection, the inspector concluded that the technical issues related to RWST leakage were addressed by the licensee plant staff in conjunction with the corporate engineering staff. After discovery of the tank leakage, a program was planned and is being implemented to provide for characterization of the cause of leakage, assessment of the operability of the leaking tank until the next outage when repairs can be made, and providing for an action plan should continued trending of the leakage approach rates which require consideration of plant shutdown. The following actions are being taken:

(1) Monitoring of the leak rate with measurements taken daily such that any rapid change could be anticipated prior to excessive water level reduction in the RWST.

(2) Evaluation of submerged ultrasonic inspection techniques to provide for an assessment of the possibility of rapid leakage increase due to wastage corrosion or extensive corrosion cracking.

(3) Continued close cooperation with NUSCO engineering staff in evaluation of the cause and correction of the tank leakage to provide for appropriate technical expertise toward problem solution.

The inspector found these actions to be appropriate.

7. SAFETY ASSESSMENT AND QUALITY VERIFICATION (40500, 71707, 90712 and 92700)

7.1 PLANT OPERATIONS REVIEW COMMITTEE

The inspectors attended several plant operations review committee (PORC) meetings. Technical specification 6.5 requirements for required member attendance were verified. The meeting agendas included procedural changes, proposed changes to the technical specifications, plant design change records, and minutes from previous meetings. PORC meetings were characterized by frank discussions and questioning of the proposed changes. In particular, consideration was given to assure clarity and consistency among procedures. Items for which adequate review time was not available were postponed to allow committee members time for further review and comment. Dissenting opinions were encouraged and resolved to the satisfaction of the committee prior to approval. The inspectors observed that the PORC adequately monitors and evaluates plant performance and conducts a thorough self-assessment of plant activities and programs.

7.2 REVIEW OF WRITTEN REPORTS

Periodic reports and licensee event reports (LERs) were reviewed for clarity, validity, accuracy of the root cause and safety significance description, and adequacy of corrective action. The inspectors determined whether further information was required. The inspectors also verified that the reporting requirements of 10 CFR 50.73, station administrative and operating procedures, and technical specification 6.9 had been met. The following reports were reviewed:

LER 90-30 Failure to Establish Fire Watch for Inoperable Fire Door

LER 90-31 Surveillance Frequency Exceeded for Fire Penetration Seals

LER 90-32 Load Reduction Due to Inoperability of Containment Fan Coolers

Haddam Neck Plant Monthly Operation Report 90-12, covering the period December 1, 1990 to December 31, 1990

The reports reviewed contained accurate information.

7.2.1 FAILURE TO ESTABLISH FIRE WATCH

LER 90-30 describes the circumstances surrounding a failure to establish a fire watch for an inoperable fire door on November 25, 1990. The central alarm station door was inoperable for a period of about one and one half hours without the required fire watch. An operator identified this error and a fire watch was immediately posted. The root cause determination concluded that personnel error resulted in untimely establishment of the fire watch. The personnel involved were counseled and operator training will be enhanced to preclude further similar incidents.

Technical Specifications (TS) 3.7.7 requires that a fire watch be posted within one hour to compensate for an inoperable fire door. Although this event constitutes a violation of TS, the criteria of section V.G.1 of 10 CFR Part 2, Appendix C, Enforcement Policy (1990) were met, and no Notice of Violation will be issued **(50-213/90-20-01)**.

7.2.2 FIRE BARRIER SURVEILLANCE FREQUENCY EXCEEDED

On November 30, 1990, engineering personnel determined that the 18-month surveillance frequency for inspection of fire barrier seals had been exceeded for the period between May 7, 1989 and April 4, 1990. The inspection of fire barrier seals was being conducted in response to NRC Information Notice 88-04, "Inadequate Qualification and Documentation of Fire Barrier Penetration Seals". A penetration seal upgrade program was established in 1989 and implemented in 1989 and 1990. Apparently, when this program was planned, the licensee overlooked the TS 4.15.F.1 (a) requirement for fire penetration seal inspection. The inspection was required to be complete by May 7, 1989 but was not completed until April 4, 1990. Although this event constitutes a violation of TS, the criteria of section V.G.1 of 10 CFR Part 2, Appendix C, Enforcement Policy (1990) were met, and no Notice of Violation will be issued **(50-213/90-20-02)**.

8. EXIT INTERVIEWS

During this inspection, periodic meetings were held with station management to discuss inspection observations and findings. At the close of the inspection period, an exit meeting was held to summarize the conclusions of the inspection. No written material was given to the licensee and no proprietary information related to this inspection was identified.

In addition to the exit meeting for the routine resident inspection, the following meetings were held for inspections conducted by Region I based inspectors.

Report No.	Inspection Dates	Reporting Inspector	Areas Inspected
213/91-01	Jan. 7 - Jan. 11	S. Sherbini	Radiation Controls
213/91-02	Jan. 7 - Jan. 11	J. Jang	Environmental Monitoring and Effluents
213/91-03	Jan. 22 - Jan. 24	A. Finkel	ATWS Follow-up

8.1 MANAGEMENT MEETING

A meeting was held with licensee management representatives at the NRC Region I office on January 22, 1991 to discuss the licensee's programs for self-assessment. Attachment No. 1 to this report is a list of attendees at this meeting. The slides used by the licensee to supplement the presentation are contained in Attachment 2.

An initial meeting was held on site on December 19 following inspector review of the self-assessment program. This was addressed in a previous inspection (see NRC Inspection Report 50-213/90-19, section 7.4).

The presentation included discussions on recent management changes within the licensee's organization and assessments on performance within the areas of operations, radiological controls and chemistry, emergency preparedness, physical security and the fitness for duty program, engineering technical support and overall safety assessment and quality verification. In addition the results of the licensee's safety system functional inspection of the service water system were summarized.

Within these functional areas the licensee presented its assessment of program and plant performance and identified areas where improvement is desired. Emphasis was placed on training of licensed and non-licensed operators, technicians and first level supervisors. New initiatives include team training and inter-system loss-of-coolant accident training. Staffing of licensed operators to upgrade work control was discussed along with the program for quality assurance audits of operating procedures.

Within the area of radiation protection, the licensee discussed improvements made to procedures which are now part of the site radiation protection manual and also improvements in corporate support to the site. The licensee presented its assessment of performance in maintaining personnel exposure as low as reasonably achievable, reducing solid radioactive waste shipments and controlling liquid and gaseous radioactive waste effluents. Of particular interest were discussions of the licensee's exposure reduction program, new methods for tracking individual radiation exposure, appraisals made by the corporate staff of the site radiation protection program performance and experience with new personnel dosimetry equipment.

The augmented fuel clad performance monitoring program was discussed. This program was developed in response to debris-induced fuel clad damage experienced during the last operating cycle. The performance of the fuel clad and the monitoring program were included in these discussions.

The licensee indicated that its assessments of performance and recent industry initiatives had caused changes to be made within the organizations and programs which support plant maintenance and surveillance. These included additional supervisory positions and new initiatives in training. Job task analysis has been introduced into new training programs. Containment performance was addressed in terms of the results of the integrated and local leak rate tests along with programs considered for the future.

Initiatives in the area of emergency preparedness include the use of the control room simulator for the 1991 exercise and training drills. Because the simulator is located adjacent to the Millstone Site, its use as the basis for the exercise scenario presents significant logistic problems.

Program improvements to encourage personnel retention within the contract security force were addressed along with improvements in equipment and programs. The performance of the licensee's fitness for duty program was discussed based on recently identified deficiencies in the required supervisor training program (NRC Inspection Reports 50-213/90-17 and 90-18).

Within the area of engineering and technical support, the licensee summarized the status of major projects completed including modernization of the reactor protection system, replacement of the nuclear instrumentation, completion of the new electrical switchgear room, removal of the reactor vessel thermal shield, and implementation of revised standard format technical specifications and the service water system. The results of over three years of engineering review of the service water system were discussed. The conclusions of the safety system functional inspection and proposed improvements to the system were described.

The licensee identified future efforts in improving the auxiliary feedwater system, the motor-operated valve monitoring program, the check valve program, and the service water system.

Projects currently scheduled for completion during operating cycle No. 16 and the Fall, 1991 refueling outage include the replacement of the reactor neutron flux mapping system, replacement of the security computer and its communication equipment, modifications to improve the control of personnel radiation exposure, and replacement of the screen house water hypochlorite system. Modifications will be made to the auxiliary feedwater system turbine steam admission sub-system, to provide independence from instrument air and operator interface during a system automatic initiation, and turbine governor with a modern microprocessor-based governor system. Additionally, instrumentation to support more reliable reactor vessel level indication during mid-loop operation is scheduled for installation.

Operating cycle No. 17 is expected to include the modernization of the steam generator feedwater control system, and the relocation of diesel generator and motor operated valve controls from a control room back panel to the main control board.

The licensee also addressed the advances made in the reduction of the probability for core melt through the implementation of recommendations made by a twelve member multi-disciplined task force.

The NRC staff and management found this transfer of information to be helpful and useful.

ATTACHMENT 1

The following is a list of attendees at the Licensee Initiatives and Self Assessment presentation in the NRC Region I Office on January 22, 1991.

Licensee Attendees:

W. Romberg, Vice President, Nuclear Operations
J. Stetz, Nuclear Station Director
G. Bouchard, Nuclear Unit Director
D. Ray, Nuclear Services Director
G. Johnson, Director Generation Engineering and Design
R. Rogers, Manager, Radiological Assessment Branch
S. Thickman, Supervisor, Nuclear Safety Engineering
G. VanNoordennen, Supervisor Nuclear Licensing

NRC Attendees:

W. Kane, Deputy Regional Administrator
L. Bettenhausen, Chief, Operations Branch, DRS
J. Joyner, Chief, Facilities Radiological Safety and Safeguards Branch, DRSS
J. Linville, Chief, Projects Branch No. 1, DRP
W. Pasciak, Chief, Facilities Radiation Protection Section, DRSS
K. Brockman, Regional Coordinator, OEDO
D. Haverkamp, Chief, Reactor Projects Section 4A, DRP
J. Shedlosky, Senior Resident Inspector
A. Wang, Project Manager, PD 1-4, NRR

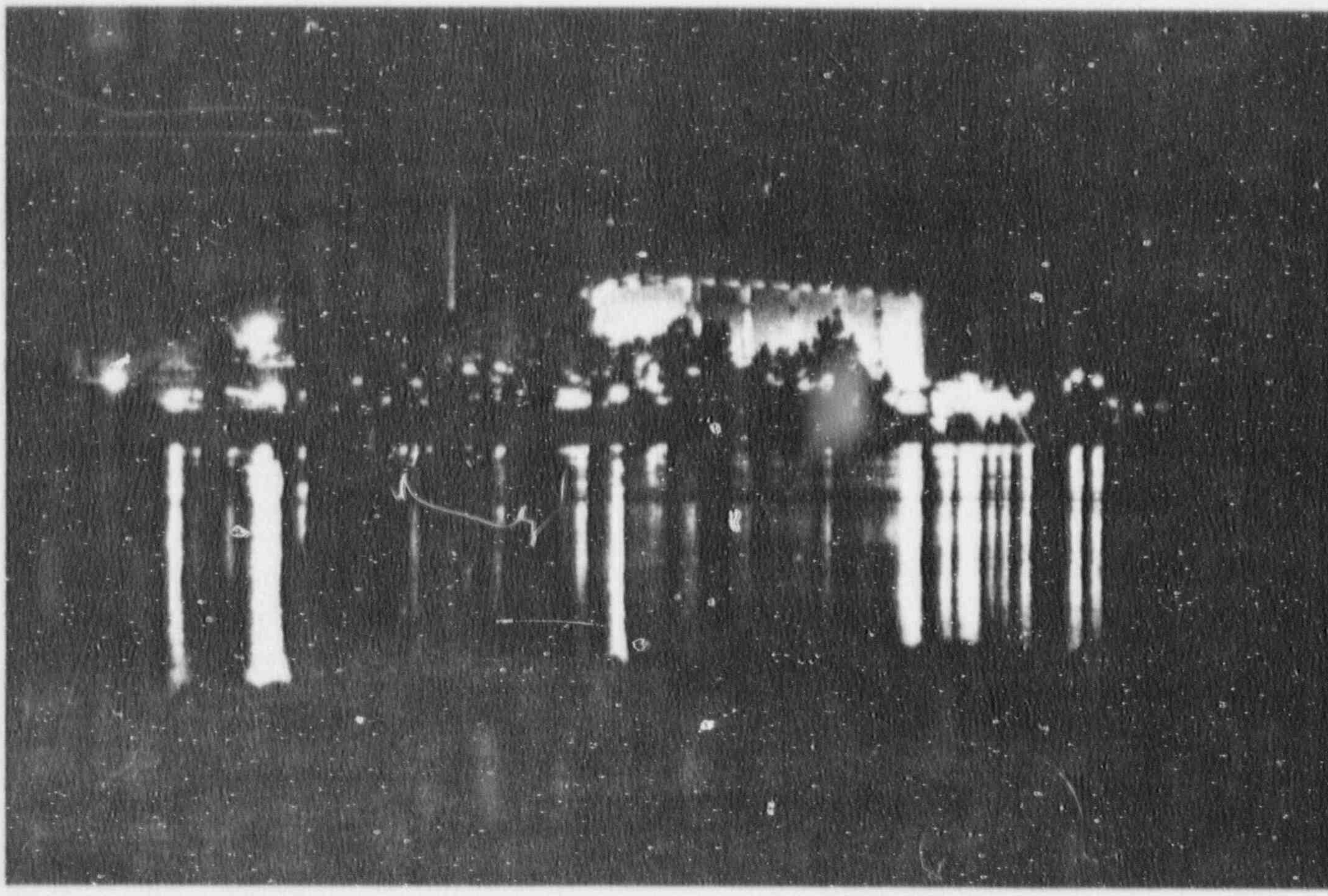
Guests:

A. Demyanenko, Chief, GPAN Southwest Region, USSR
V. Koltonov, Senior Resident Inspector, GPAN, USSR
A. Gluchov, Inspector, GPAN, USSR
N. Berkoff, Interpreter, U.S. Department of State

ATTACHMENT 2

JANUARY 21, 1991 MEETING SLIDES

CONNECTICUT YANKEE ATOMIC POWER CO.



Mid Salp Report

January 1991

HADDAM NECK MID-SALP CYCLE SELF ASSESSMENT

U. S. NUCLEAR REGULATORY COMMISSION

JANUARY 22, 1991

NORTHEAST UTILTIES

ATTENDEES

Wayne D. Romberg	Vice President, Nuclear Operations
John P. Stetz	Nuclear Station Director
Gary H. Bouchard	Nuclear Unit Director
Donald J. Ray	Nuclear Services Director
G. Leonard Johnson	Director, Generation Engineering and Design
Reginald C. Rodgers	Manager, Radiological Assessment Branch
Stuart A. Thickman	Supervisor, Nuclear Safety Engineering
Gerard P. Van Noordenen	Supervisor, Nuclear Licensing

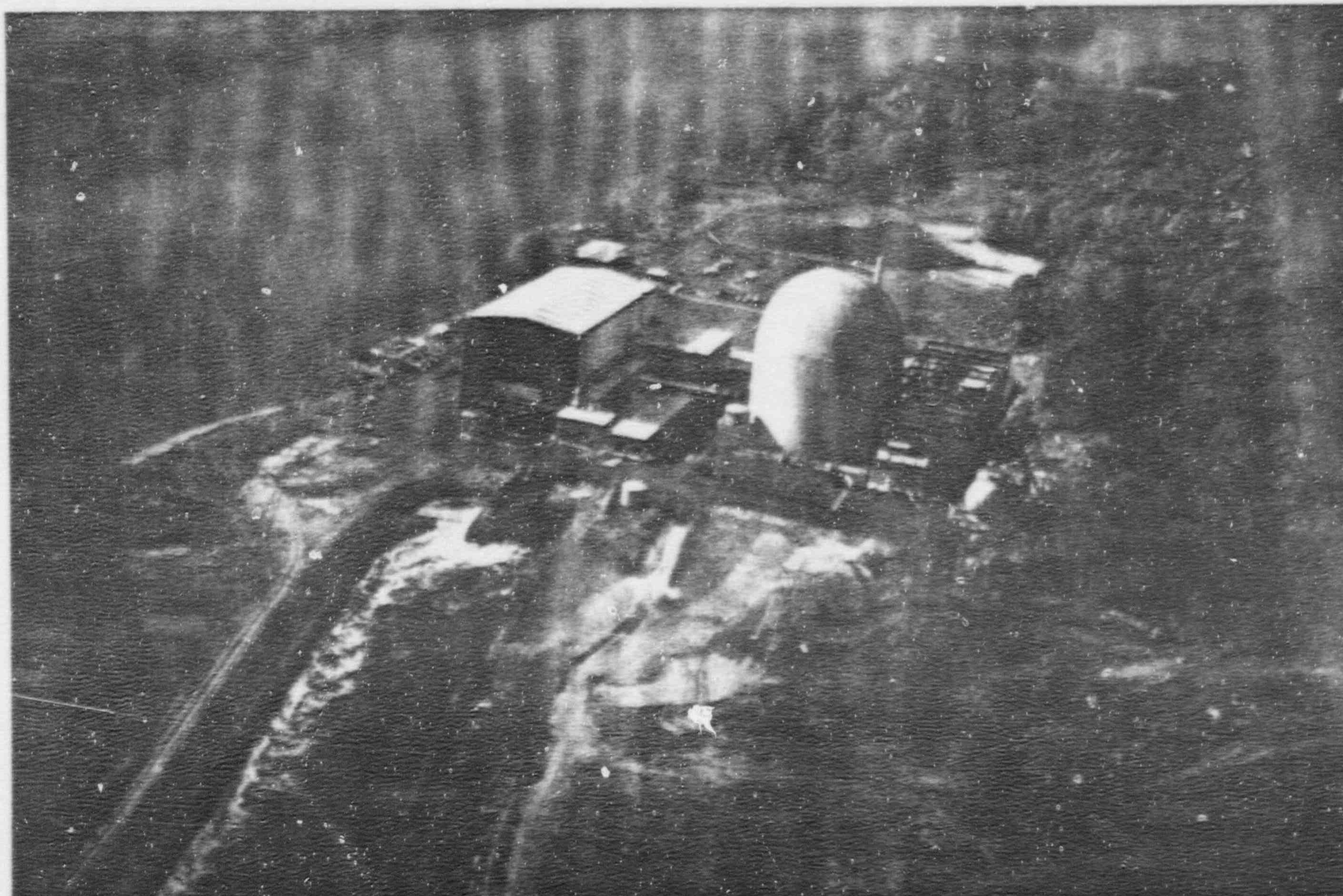
AGENDA

Opening Remarks	W. D. Romberg
Station Organization Changes	J. P. Stetz
Status of NU Initiatives in SALP Functional Areas	
Operations	G. H. Bouchard
Radiological Control/Chemistry	D. J. Ray/R. C. Rodgers
Maintenance/Surveillance	G. H. Bouchard
Emergency Preparedness	D. J. Ray/R. C. Rodgers
Security/Safeguards	D. J. Ray
Engineering/Technical Support	G. L. Johnson/G. H. Bouchard
Safety Assessment/Quality Verification	J. P. Stetz
Results of Service Water SSFI	S. A. Thickman
Closing Remarks	W. D. Romberg

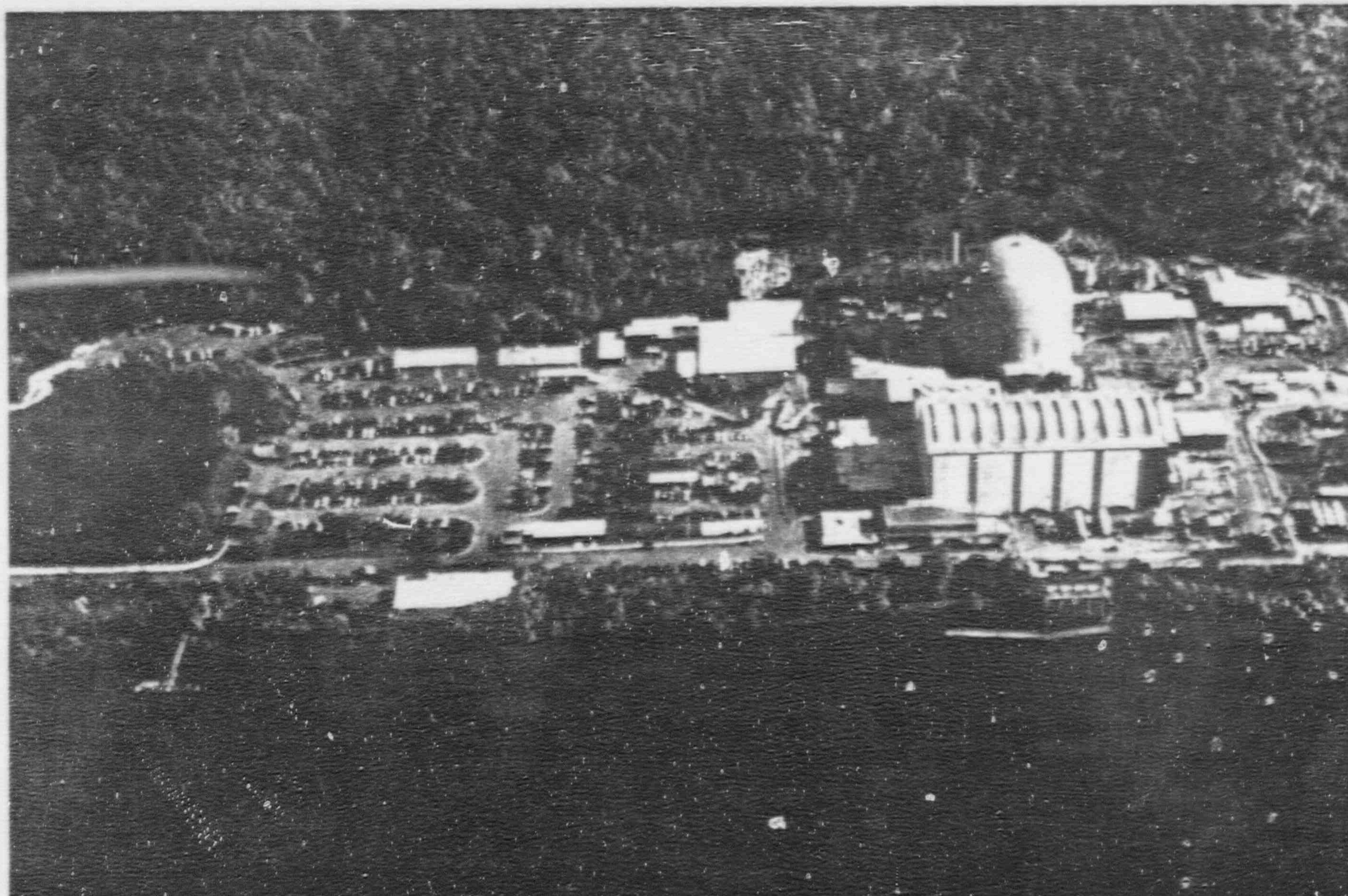
CONNECTICUT YANKEE

WAYNE D. ROMBERG

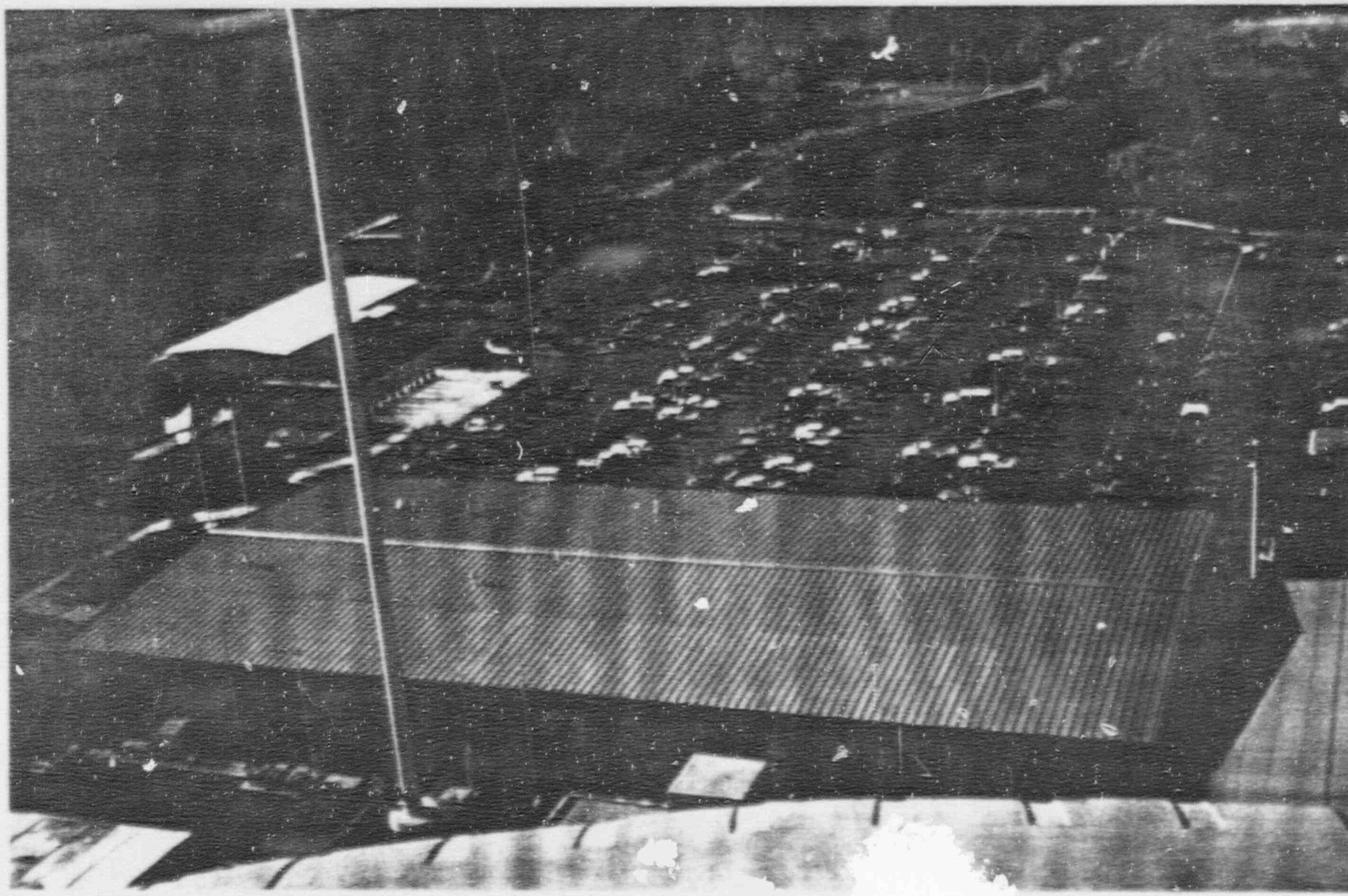
VICE PRESIDENT
NUCLEAR OPERATIONS



Connecticut Yankee - 1972



Connecticut Yankee - 1987

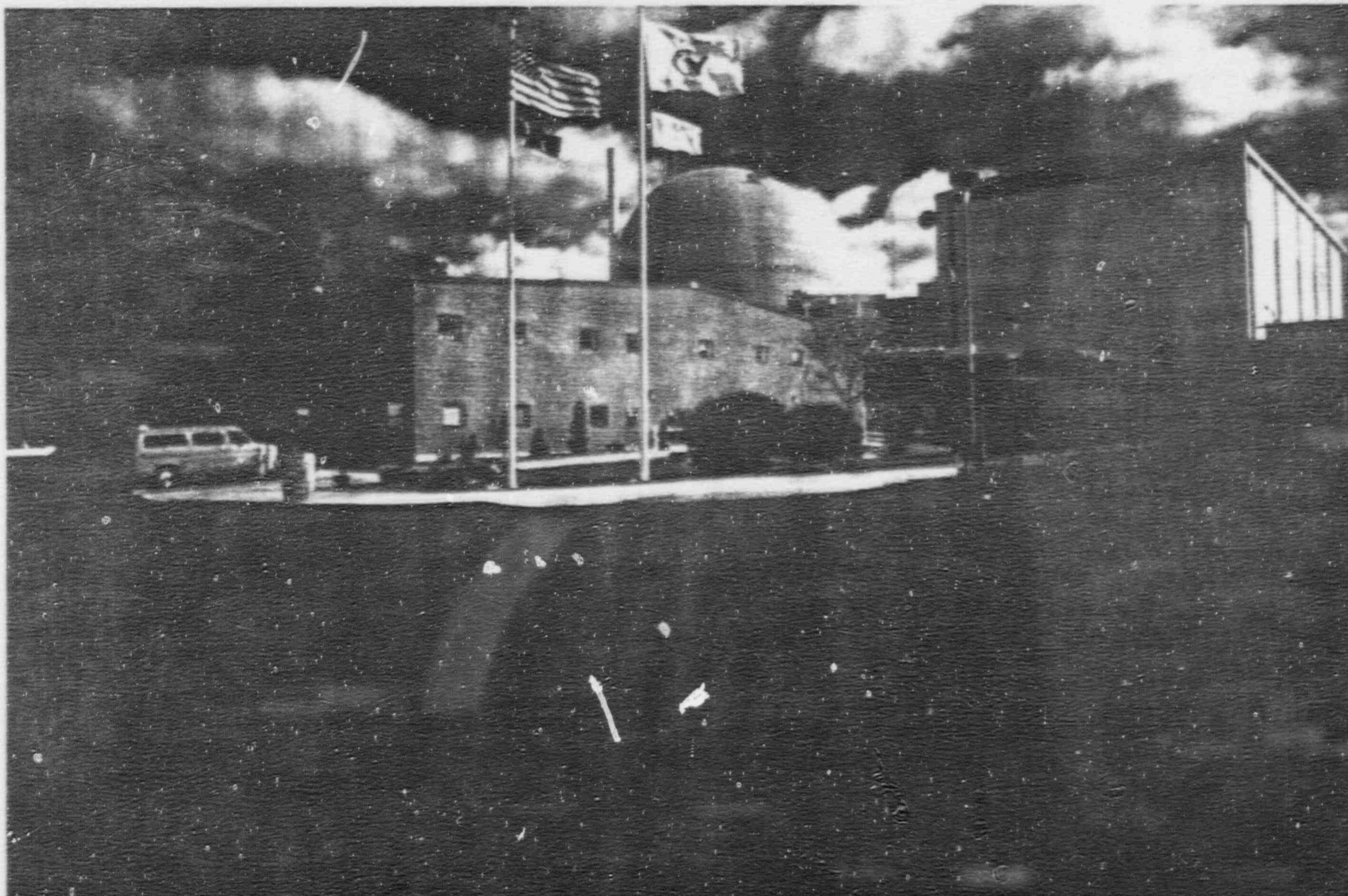


Connecticut Yankee - 1990

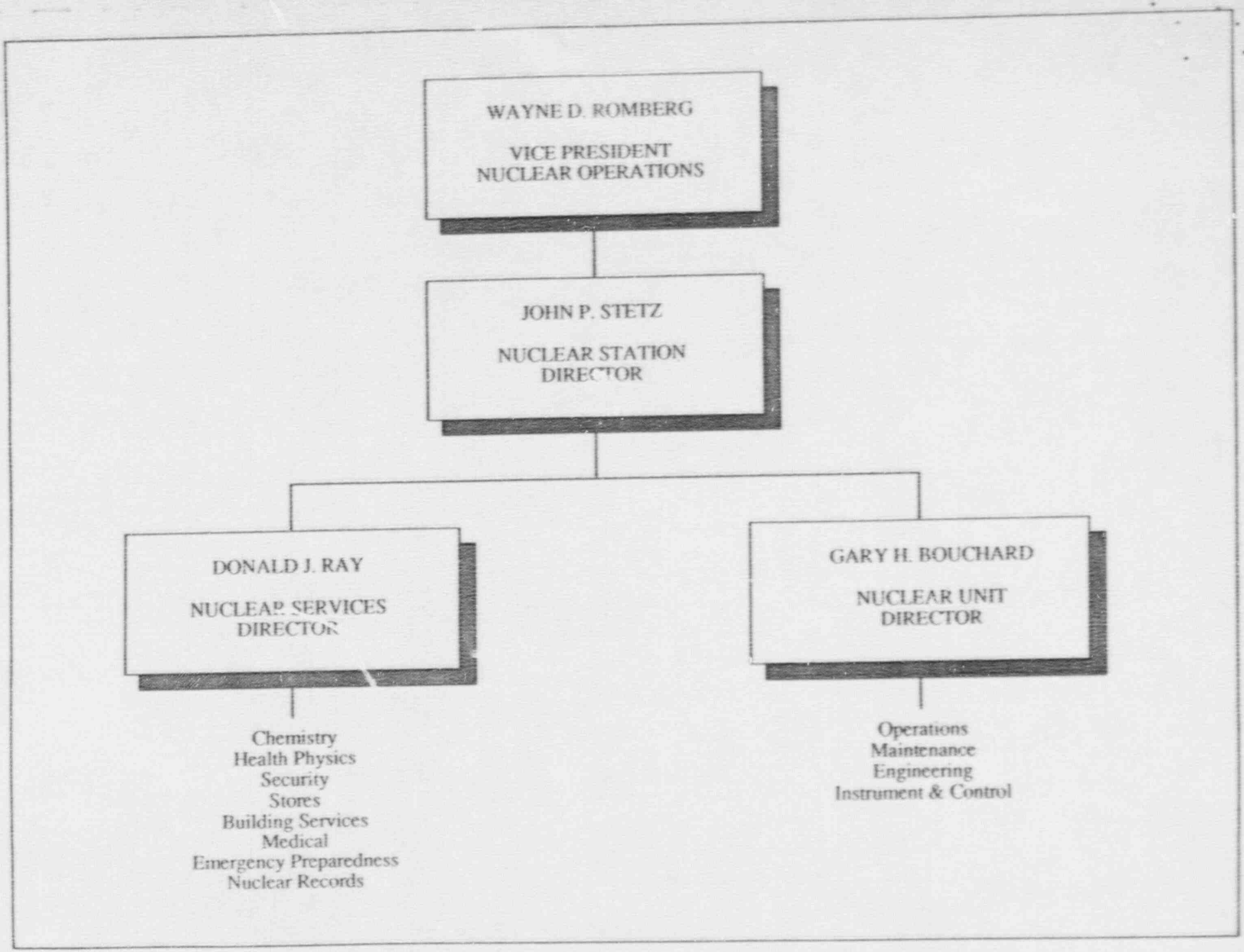
CONNECTICUT YANKEE

JOHN P. STETZ

NUCLEAR STATION DIRECTOR



Connecticut Yankee - 1990



WAYNE D. ROMBERG
VICE PRESIDENT
NUCLEAR OPERATIONS
JOHN P. STETZ
NUCLEAR STATION
DIRECTOR
DONALD J. RAY
NUCLEAR SERVICES
DIRECTOR
GARY H. BOUCHARD
NUCLEAR UNIT
DIRECTOR
Chemistry
Health Physics
Security
Stores
Building Services
Medical
Emergency Preparedness
Nuclear Records
Operations
Maintenance
Engineering
Instrument & Control

STATION ORGANIZATIONAL CHANGES

- Station Management
  - Station Director
  - Nuclear Services Director

- Department Managers
  - Maintenance
  - Operations
  - Instrument & Control
  - Security

CONNECTICUT YANKEE

GARY H. BOUCHARD

NUCLEAR UNIT DIRECTOR

Operations
Maintenance
Engineering
Instrument & Control

OPERATIONS

Station Status

Operator Requalification Training

Licensed Operator Initial Training

Non-Licensed Operator Training

Procedures

Staffing

PORC

OPERATIONS

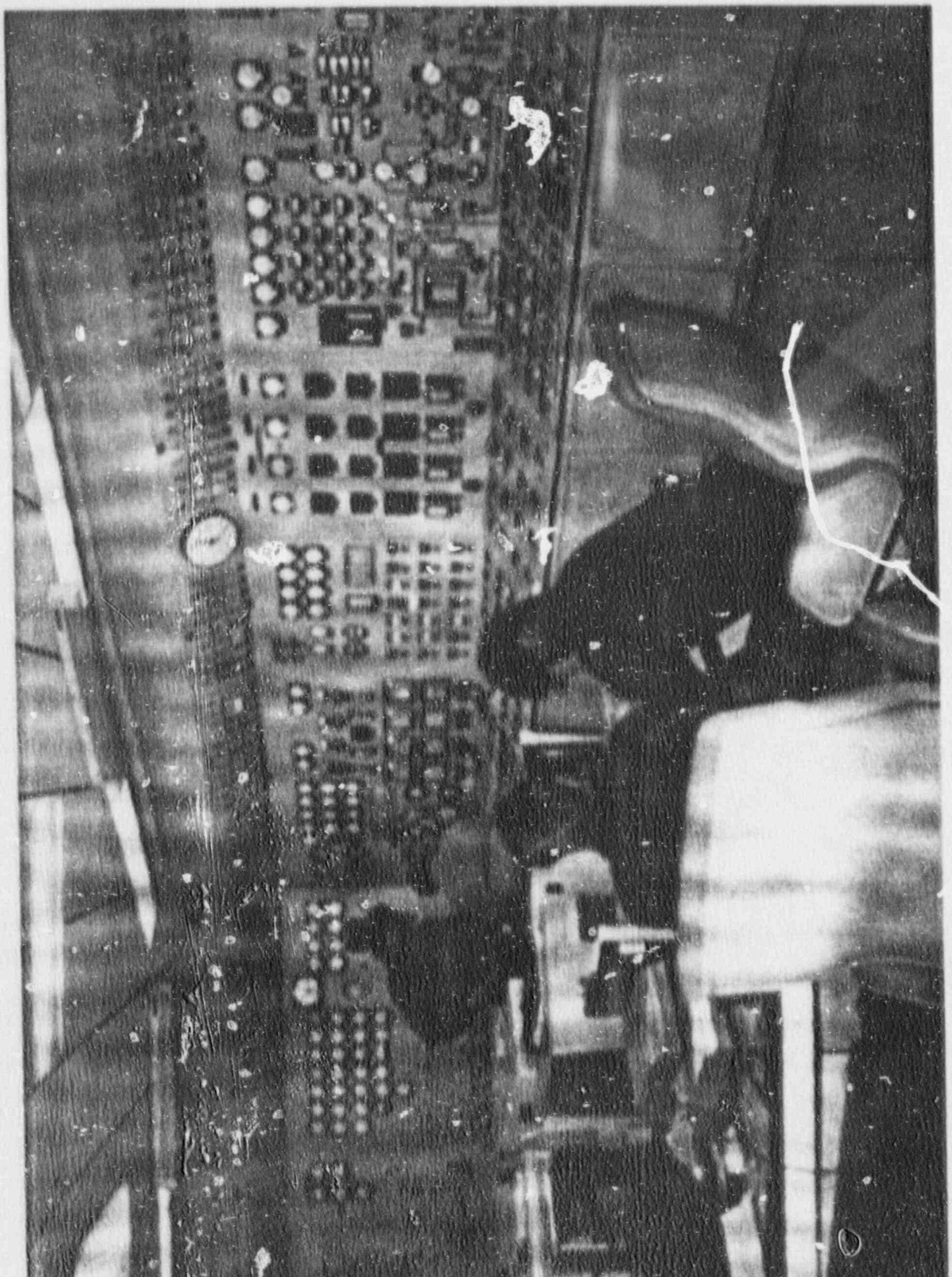
September 1989 - August 1990 Refueling Maintenance Outage

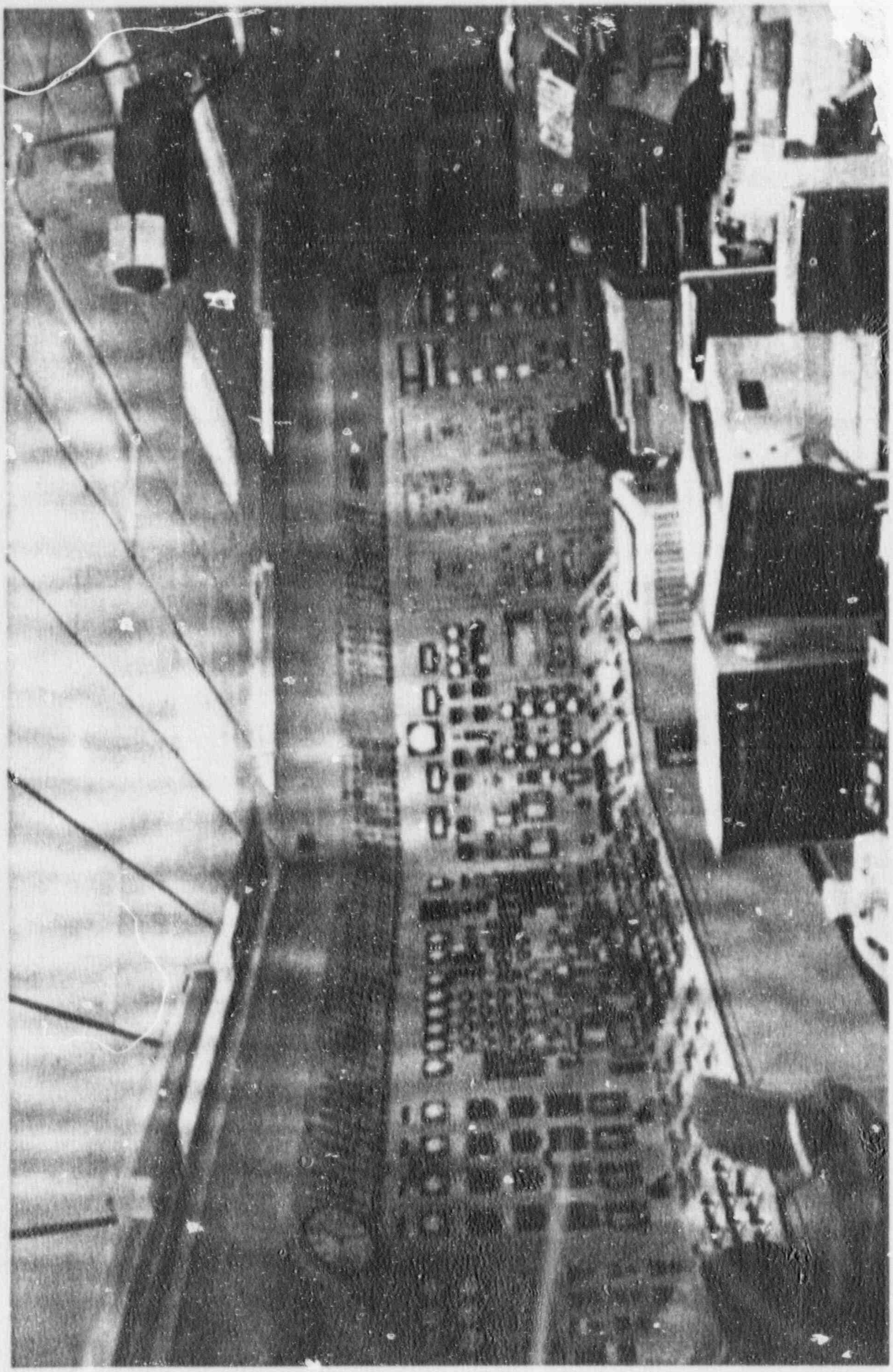
September 1-8, 1990 Feedwater Regulating Valve Problem

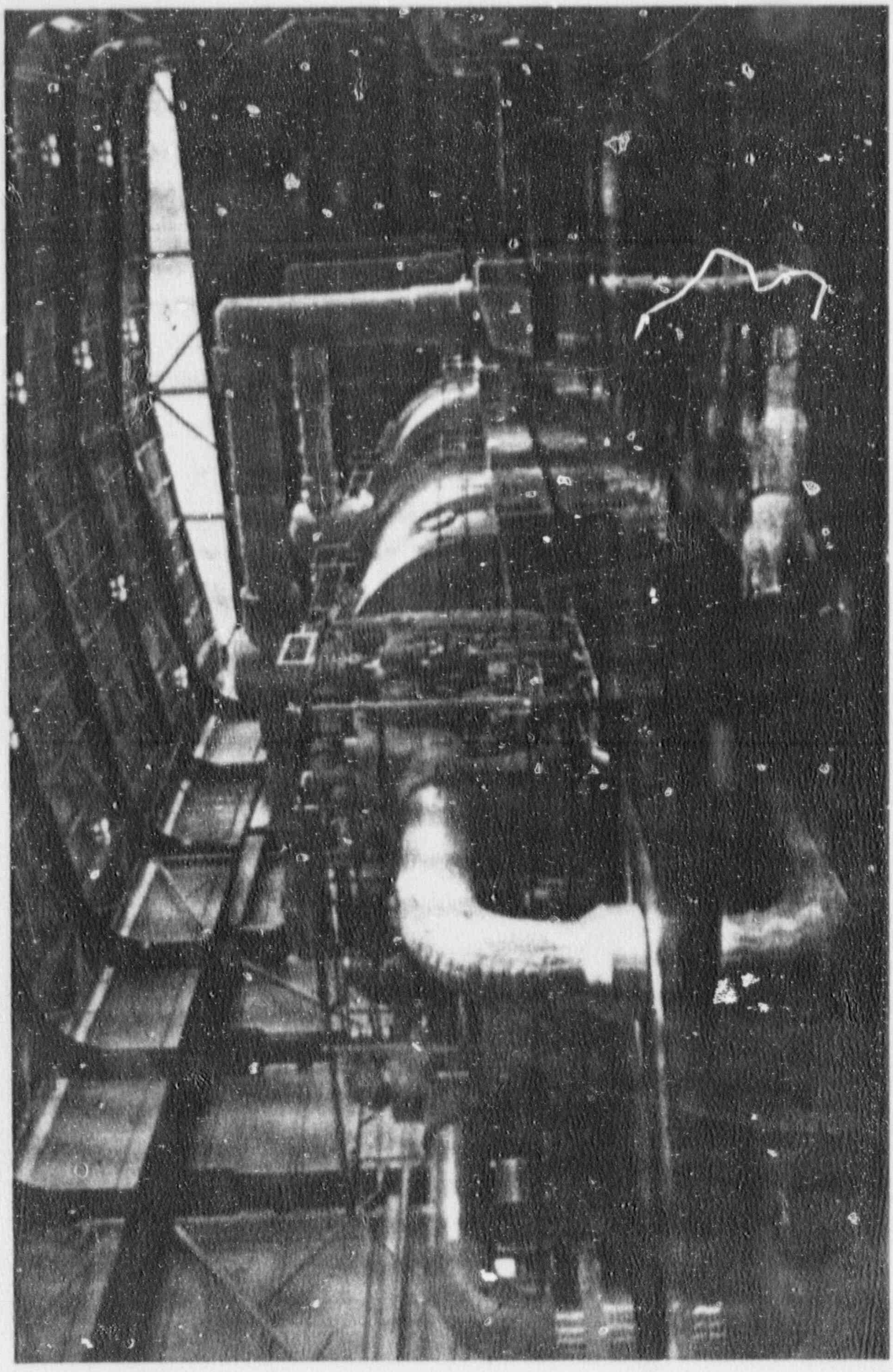
September 20-26, 1990 Condensate Pump Suction Boot

October 27 - November 15, 1990
Containment Air Recirculation Fan Cooler Cleaning

Current Station Status







OPERATOR REQUALIFICATION TRAINING

- Emphasis on Quality of Requalification Program
  - New EOP Flow Charts
- Team Training
  - Psychology of Teamwork
  - Hands On Skills Exercises
- ISLOCA Training
  - RHR & SI Simulator Software Being Remodeled
  - Scheduled to be Operational 1st Quarter 1991
  - 3 Scenarios
    - Failure HPSI Check Valves
    - Failure LPSI Check Valves
    - RHR Suction Pipe Rupture in PAB
  - Training on a 3 Year Cycle After 1st Quarter 1991

CONNECTICUT YANKEE EMERGENCY ACTION LEVELS — EVENT BASED TABLE

GENERAL EMERGENCY BRAVO · S[illegible]ENCY CHARLIE-TWO · ALERT CHARLIE-ONE · UNUSUAL EVENT DELTA-ONE

SECURITY THREAT — Symptom	SECURITY THREAT — Condition	FIRE — Symptom	FIRE — Condition	NATURAL PHENOMENON — Symptom	NATURAL PHENOMENON — Condition	MISCELLANEOUS EVENTS — Symptom	MISCELLANEOUS EVENTS — Condition	CLASSIFICATION / POSTURE CODE
PHYSICAL ATTACK: [illegible]		[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	GENERAL EMERGENCY
PHYSICAL ATTACK: Physical plant attack WITH imminent occupance of CR OR other Vital Areas described in the security procedures.	Imminent loss of physical control of the plant.	FIRE SAFETY SYSTEM: OSS opinion OR visual observation of fire compromising the functions of safety systems.	Fire compromising the functions of safety systems.	EARTHQUAKE/STORM: Plant not in cold S/D WITH observation OR notification of earthquake > SSE Spectra (0.17g ZPA) OR tornado striking plant OR sustained winds > 85 mph onsite OR flood (> [illegible] ft.), low water or surge > design basis level.	Natural phenomenon being experienced OR forecast to be > design basis level while plant NOT in cold shutdown.	HAZARDS: Observation of: [illegible]. CR EVACUATION: Evacuation of CR AND control of S/D system NOT established from local S/D stations in 15 min.	[illegible] OR [illegible]. CR EVACUATION: [illegible]	SITE AREA EMERGENCY — CHARLIE-TWO — Events in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public.
SECURITY COMPROMISE: Ongoing security compromise (e.g., an actual entry by unauthorized person or persons into the Protected Area).	Ongoing security compromise.	FIRE SAFETY SYSTEM: OSS opinion OR visual observation of fire potentially affecting the safety systems.	Fire potentially affecting safety systems.	EARTHQUAKE/STORM: Observation or notice of earthquake > OBE Spectra (0.09g ZPA) OR tornado striking facility [illegible] OR sustained winds > 85 mph onsite OR flood [illegible] near design basis level.	Natural phenomenon being experienced OR forecast near design basis level.	HAZARDS: Observation of: [illegible]. CR EVACUATION: Anticipated or required evac of CR WITH control of S/D systems from local stations.	Other hazards being experienced OR projected [illegible] NOT [illegible] cold S/D. CR EVACUATION: Anticipated or required evac of CR WITH control of S/D systems from local stations.	ALERT — CHARLIE-ONE — Events in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant.
SECURITY THREAT: [illegible] OR [illegible] OR [illegible]	Attempted [illegible] entry OR [illegible] OR [illegible]	FIRE > 10 MIN: Visual observation of fire lasting > 10 min within the plant's Protected Area.	Fire lasting > 10 min. within the plant's Protected Area.	EARTHQUAKE/STORM: Direct observation OR [illegible] OR [illegible] OR [illegible]	Natural phenomenon being experienced or projected [illegible] OR [illegible] OR [illegible] OR [illegible]	HAZARDS: [illegible]. DESIGN LIMITS: [illegible]. CONTAMINATED INDIVIDUAL: [illegible]. OTHER: [illegible]	[illegible]. DESIGN LIMITS: [illegible] OR [illegible]. CONTAMINATED INDIVIDUAL: [illegible]. OTHER: [illegible]	UNUSUAL EVENT — DELTA-ONE — Events in progress or have occurred which indicate a potential degradation of the level of safety of the plant.

CONNECTICUT YANKEE EMERGENCY ACTION LEVELS
BARRIER FAILURE REFERENCE TABLE
THRESHOLDS FOR LOSS OR POTENTIAL LOSS OF BARRIERS

IMMINENT LOSS corresponds to the loss or potential loss of a barrier within 2 hours with no turnaround expected.

GENERAL EMERGENCY: Loss of two barriers **AND** a potential loss of the third **SITE AREA EMERGENCY**: Loss or potential loss of two barriers **ALERT**: Loss or potential loss of Fuel Clad **OR** RCS **UNUSUAL EVENT**: Loss or potential loss of Ctmt

A. FUEL CLAD BARRIER

1. Critical Safety Function Status

LOSS	POTENTIAL LOSS
Core Cooling — Red	Core Cooling — Orange

OR

2. Primary Coolant Concentration

LOSS	POTENTIAL LOSS
Dose rate at one foot from unpressurized RCS sample ≥ 130 mR/hr/ml	Not Applicable

OR

3. Five (5) or More CETs

LOSS	POTENTIAL LOSS
> 1200°F	700°F and increasing

OR

4. RVLIS

LOSS	POTENTIAL LOSS
Not Applicable	RVLIS plenum = 0%

OR

5. Ctmt Hi-Rad

LOSS	POTENTIAL LOSS
a. **WITHOUT** release to Ctmt: CD 1/2 > 5 R/hr = LOSS b. **WITH** release to Ctmt:	Not Applicable

Time after S/D	CD 1/2 R/hr	OR	Ctmt Hatch R/hr
0 – 0.5 hr	> 3000		> 250
0.5 – 4.0 hr	> 1300		> 130
4.0 – 12 hr	> 300		> 40
> 12 hr	> 40		> 6

OR

6. Other Conditions

Any condition in the opinion of the OSS/DSEO that indicates loss **or** potential loss of Fuel Clad barrier

B. RCS BARRIER

1. Critical Safety Function Status

LOSS	POTENTIAL LOSS
Not Applicable	RCS Integrity — Red **OR** Heat Sink — Red, and attempts to start an auxiliary or main feed pump have failed

OR

2. RCS Leak Rate

LOSS	POTENTIAL LOSS
> Normal makeup capacity	> 50 GPM

OR

3. RCS Subcooling

LOSS	POTENTIAL LOSS
< 0°F	< 32°F

OR

4. Ctmt Hi-Rad

LOSS	POTENTIAL LOSS
a. CD 1/2 > 5 R/hr or Ctmt Hatch > 0.5 R/hr **WITHOUT** Fuel Clad barrier loss b. **WITH** Fuel Clad barrier loss:	Not Applicable

Time after S/D	CD 1/2 R/hr	OR	Ctmt Hatch R/hr
0 – 0.5 hr	> 3000		> 250
0.5 – 4.0 hr	> 1300		> 130
4.0 – 12 hr	> 300		> 40
> 12 hr	> 40		> 6

OR

5. Other Conditions

Any condition in the opinion of the OSS/DSEO that indicates loss **or** potential loss of RCS barrier

C. CTMT BARRIER

1. Critical Safety Function Status

LOSS	POTENTIAL LOSS
Not Applicable	Ctmt — Orange

OR

2. Ctmt Pressure

LOSS	POTENTIAL LOSS
Rapid unexplained decrease following initial increase	40 psig and increasing **OR** no pressure increase when expectation exists

OR

3. Ctmt Isolation

LOSS	POTENTIAL LOSS
Valve(s) not closed **and** downstream pathway to the environment exists **OR** Ctmt isolation boundary has been bypassed by RCS leakage to a lower pressure system(s). (Utilize SG side RP and area rad monitors, as appropriate.)	Not Applicable

OR

4. SG Relief/Stm Dump/Sec Side Open with Pri-Sec Leakage

LOSS	POTENTIAL LOSS
Release of sec side to atmosphere **WITH** pri-sec leak rate > tech spec	Steam Generator Tube Rupture in progress

OR

5. Core Melt Situation

LOSS	POTENTIAL LOSS
Not Applicable	Ctmt hi rad monitor reading > [illegible]

OR

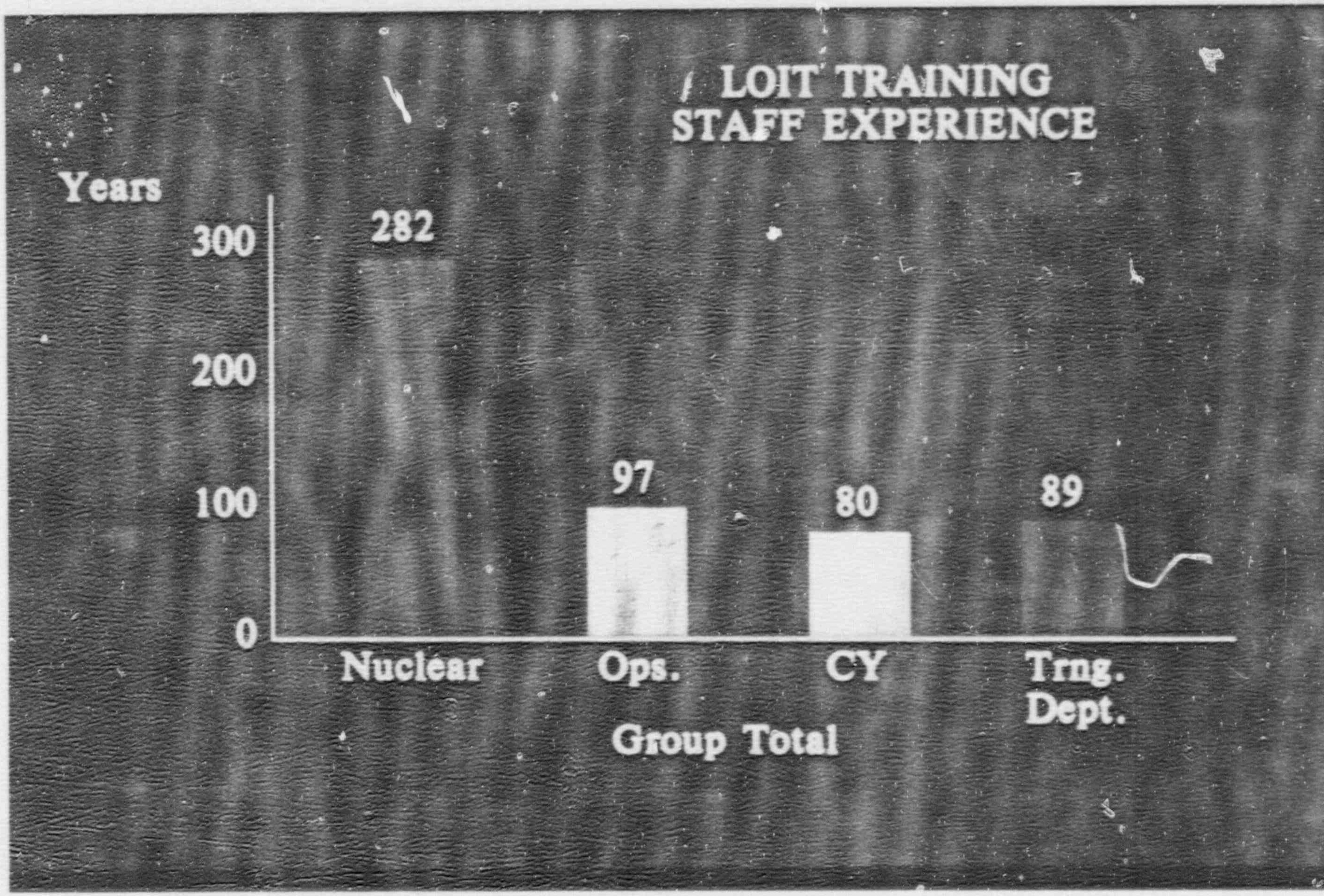
6. Radiation Releases

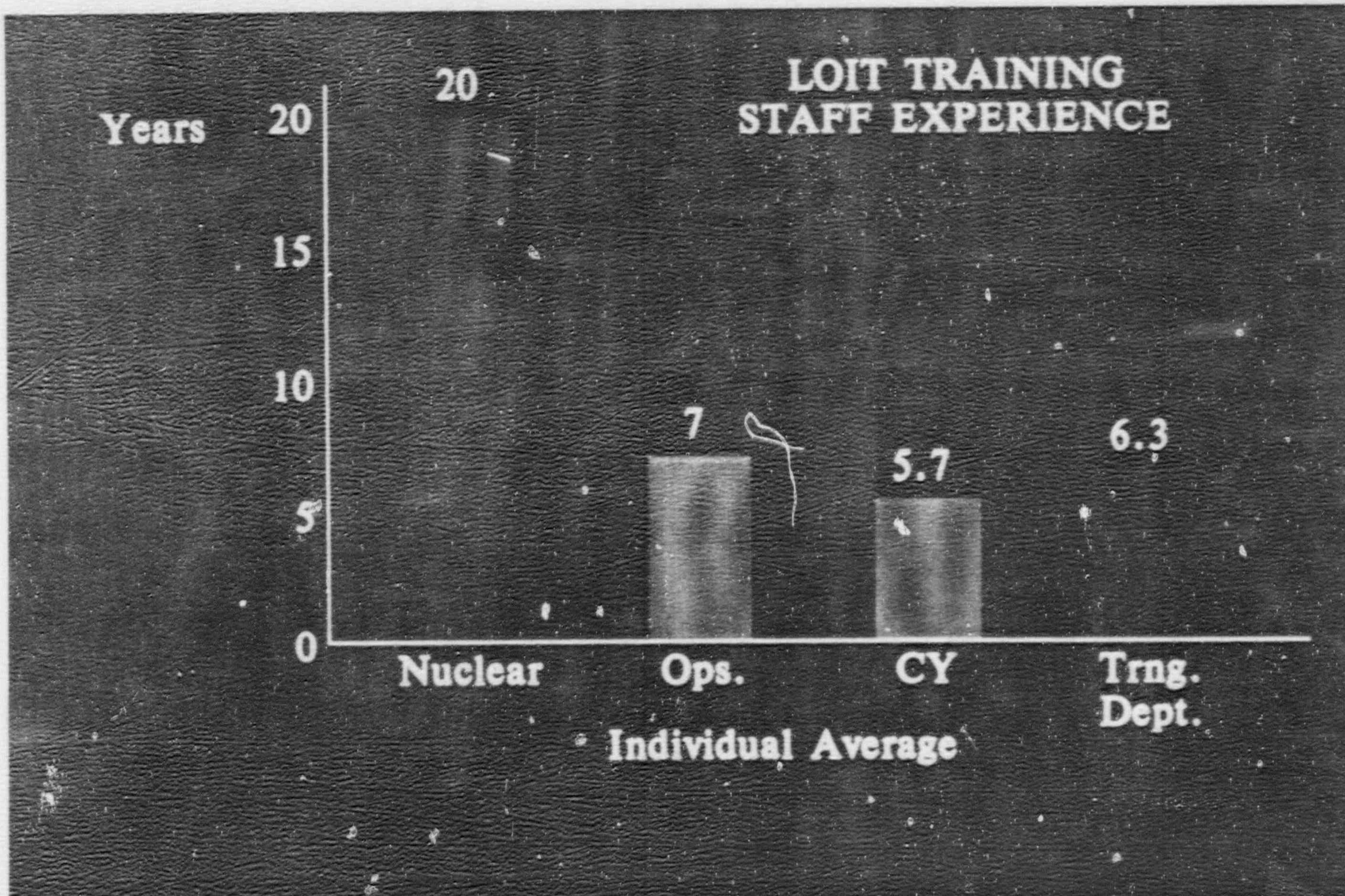
LOSS	POTENTIAL LOSS
WITH Rad Assess: Offsite plume dose rate ≥ 10^{-4} times Ctmt rad level if coolant loss is to primary Ctmt	Not Applicable

OR

7. Other Conditions

Any condition in the opinion of the OSS/DSEO that indicates loss **or** potential loss of Ctmt barrier


LOIT TRAINING
STAFF EXPERIENCE
Years
300
200
100
0
282
97
80
89
Nuclear
Ops.
CY
Trng.
Dept.
Group Total


LOIT TRAINING
STAFF EXPERIENCE
Years
20
15
10
5
0
20
7
5.7
6.3
Nuclear
Ops.
CY
Trng.
Dept.
Individual Average

NON LICENSED OPERATOR TRAINING

- Program is a Cooperative Effort
  - Shift Supervisor Involvement

- Training Program
  - AO's Qualify By Primary & Secondary Side Not By Task
  - Station Management Certifies Qualification
  - AO's Involved in Simulator Training

PROCEDURES

- EOP
    - Quality Services Department Audit Performed To Verify EOP Quality
    - Followup Audits Will Be Performed Biennially
- Other Procedures
    - Completed 2 Year Procedure Upgrade Project - AOP, NOP, ANN, ADMIN/ACP

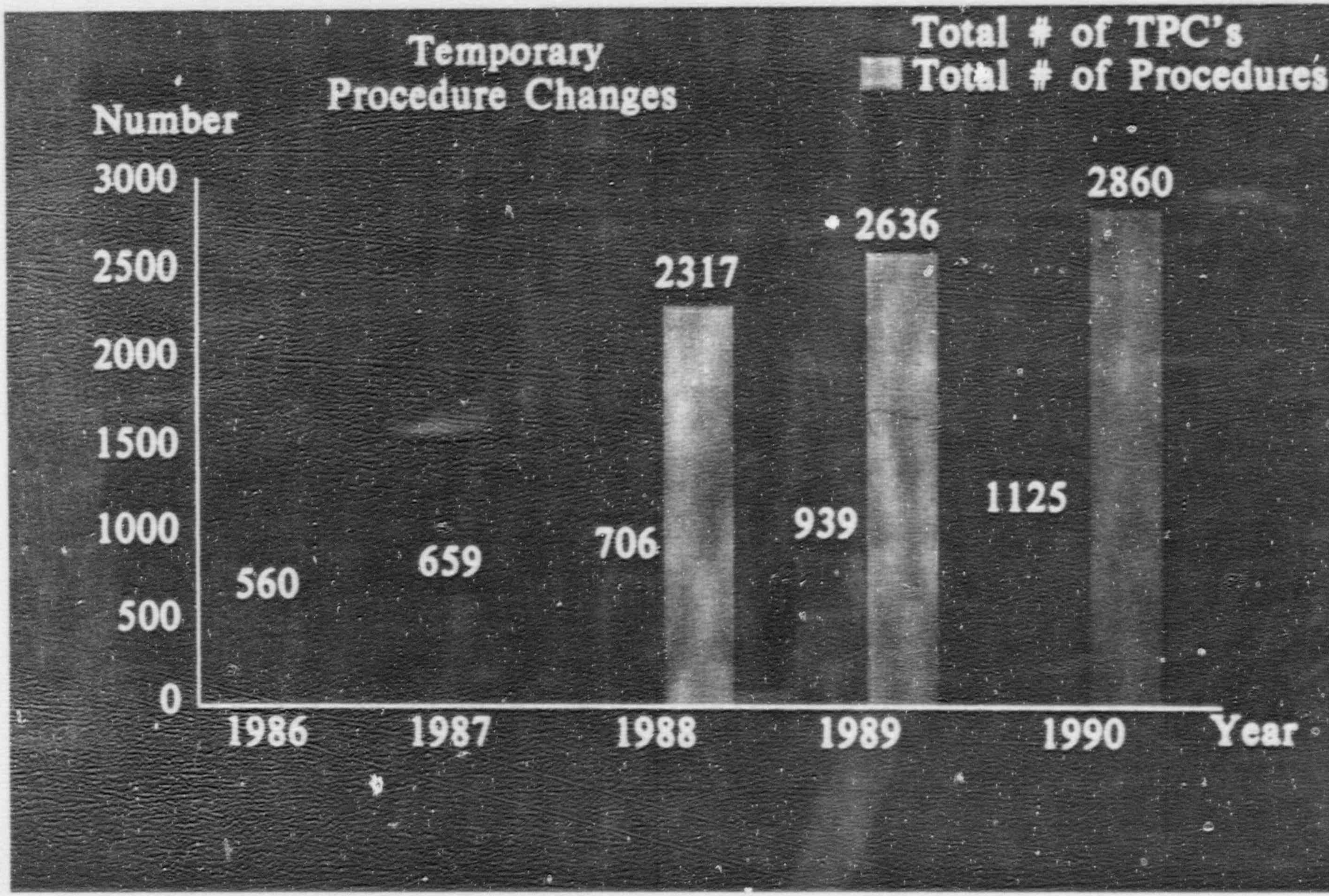
STAFFING

- Operator Experience
  - SS Average 13.5 Years
  - SCO Average 8.6 Years
  - CO Average 5 Years

- Changes
  - Adding One Auxiliary Operator
  - Upgrading Work Control
    - Shift Supervisor and CO Assigned to Work Control

PORC

- Weekly PORC Meetings
  - Ease of Access Supports Procedure Quality
  - Procedure Quality Supports Procedure Compliance
  - Temporary Procedure Changes
- Non-Routine PORCs Readily Available
- Emphasis on Quality and Critical Review of all PORC Topics
- Occasional Joint PORC/NRB Meetings on Complex Issues



Temporary
Procedure Changes
Total # of TPC's
Total # of Procedures
Number
3000
2500
2000
1500
1000
500
0
560
659
706
2317
939
2636
1125
2860
1986
1987
1988
1989
1990
Year

CONNECTICUT YANKEE

STATION SERVICES

DONALD J. RAY

NUCLEAR SERVICES DIRECTOR

CONNECTICUT YANKEE NUCLEAR SERVICES DEPARTMENTS

Health Physics	Building Services
Chemistry	Medical
Emergency Preparedness	Nuclear Records
Security	Stores

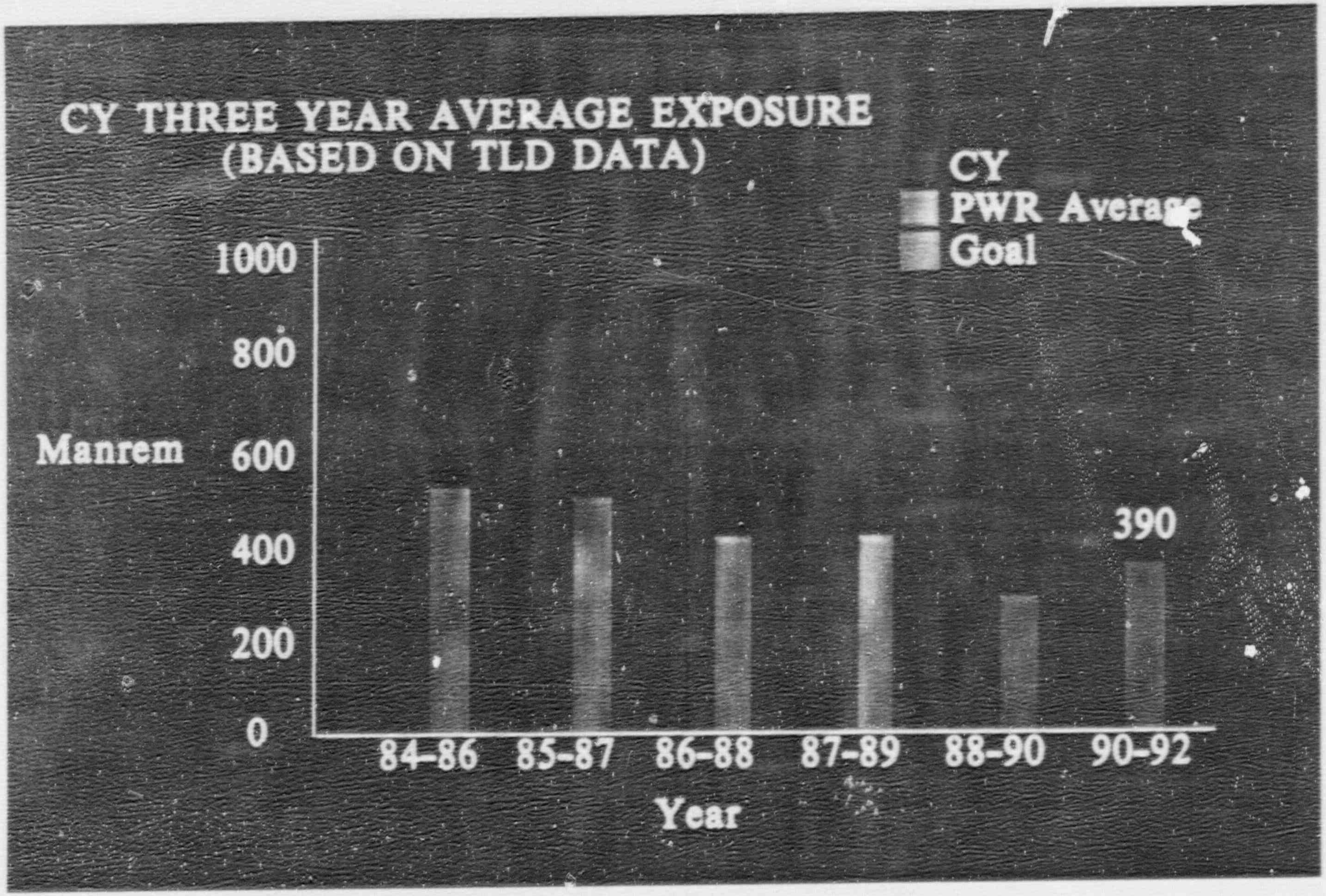
HEALTH PHYSICS

Exposure Trends

Radwaste Trends

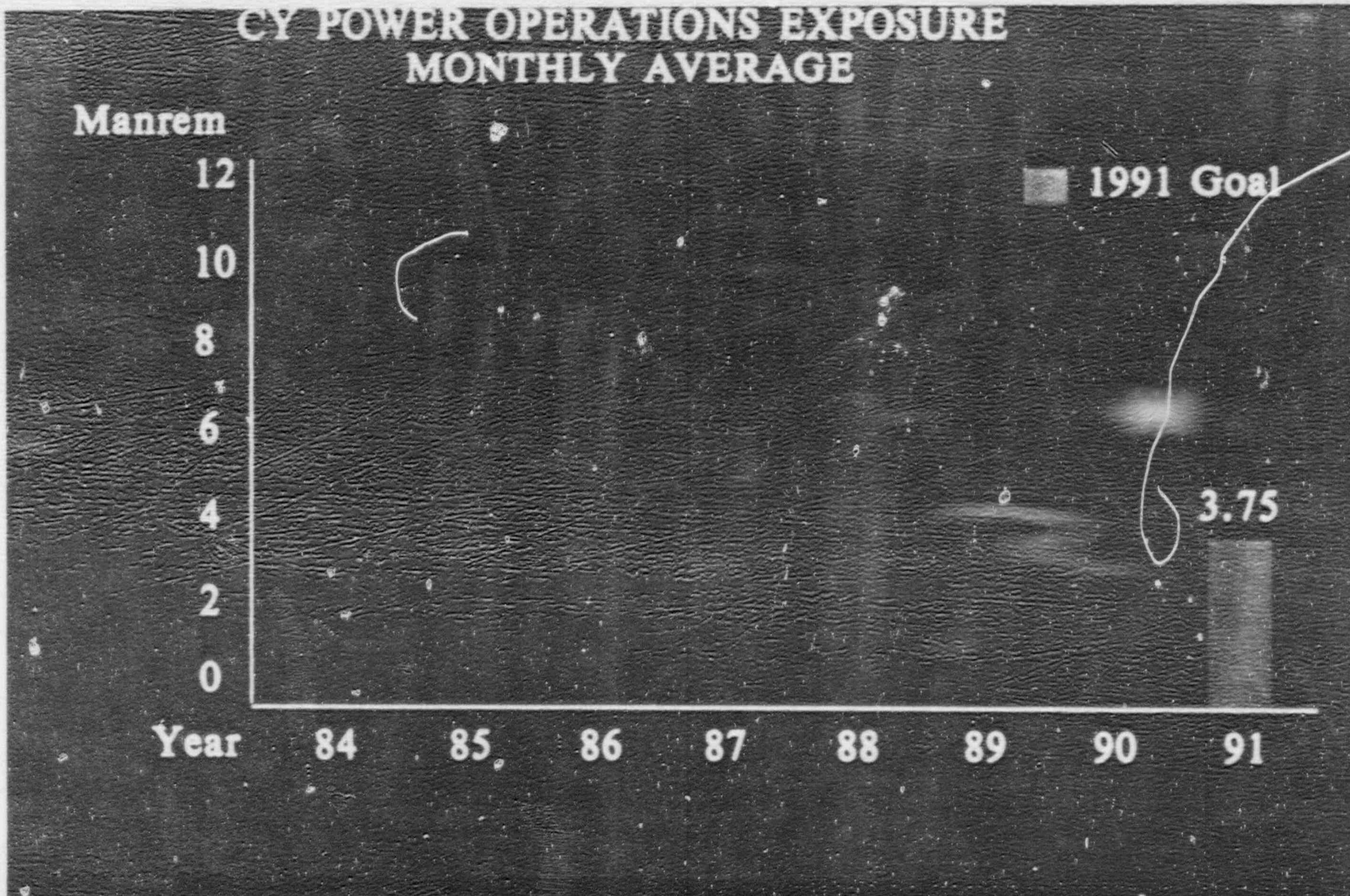
Radiation Protection Manual

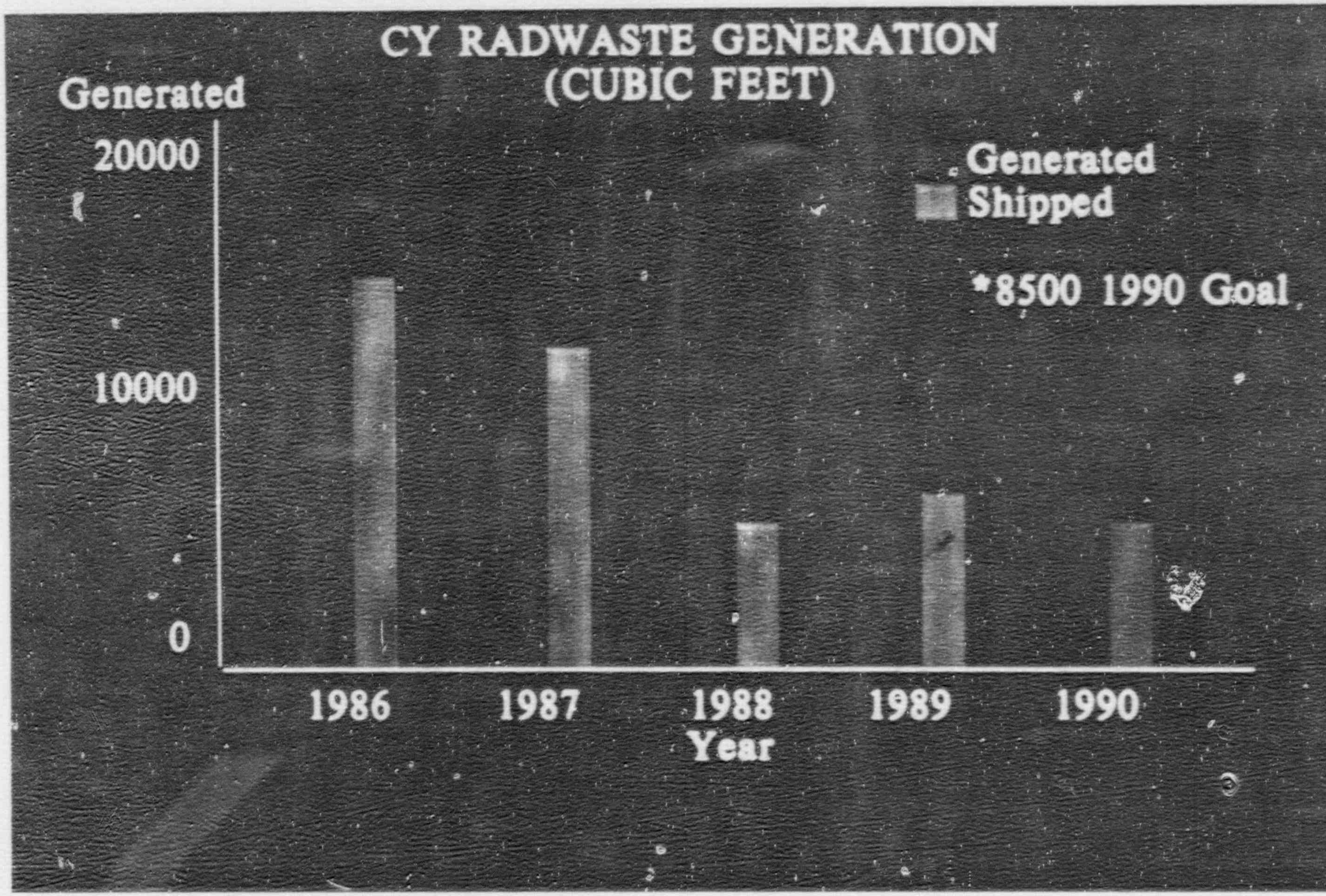
Corporate Support


CY THREE YEAR AVERAGE EXPOSURE
(BASED ON TLD DATA)
CY
PWR Average
Goal
Manrem
1000
800
600
400
200
0
390
84-86
85-87
86-88
87-89
88-90
90-92
Year

ALARA

TECHNIQUES


CY POWER OPERATIONS EXPOSURE
MONTHLY AVERAGE
Manrem
12
10
8
6
4
2
0
Year
84
85
86
87
88
89
90
91
1991 Goal
3.75


CY RADWASTE GENERATION
(CUBIC FEET)
Generated
20000
10000
0
Generated
Shipped
*8500 1990 Goal
1986
1987
1988
1989
1990
Year

RADIATION PROTECTION MANUAL

- Procedure Revision Project-Initiated in 1988
- Procedures Designed to be User Friendly
- Each Procedure Reviewed by an Assigned Technical Reviewer, Format Reviewer and Human Factors Reviewer
- Each Procedure Validated by a Primary Procedure User
- All Health Physics Procedures Have Now Been Converted to RPM Format

CONNECTICUT YANKEE

REGINALD C. RODGERS

MANAGER

Radiological Assessment Branch

RADIOLOGICAL CONTROL

Exposure Reduction Program

RITS

Health Physics Appraisals

Personnel Dosimetry Laboratory

Other Examples of Radiological Engineering Support

EXPOSURE REDUCTION PROGRAM

- ❑ Purpose to Reduce Nuclear Plant 3-Year Average Person-Rem to Industry Average/INPO Goals and Reduce Individual Lifetime Exposures
- ❑ Initiated in December 1986
- ❑ Coordinated by Radiological Assessment Branch
- ❑ Program Components
  - Goals
    - 3-Year Average Person-Rem
    - Annual Person-Rem
    - Individual Exposure
  - Exposure Reduction Initiatives
    - Short Term (1986-1990)
    - Long Term (Beyond 1990)

INDIVIDUAL EXPOSURE GOALS

- NU Employees

  - If Lifetime Exposure is Greater Than Age in Years, Limit Exposure to a Total Of:

    1990 -- 2 Rem/Year

    1991 - 1992 -- 1 Rem/Year

  - Total Exposure:

    1990 -- 3 Rem/Year

    1991 - 1992 -- 2 Rem/Year From All Sites

INDIVIDUAL EXPOSURE GOALS

- Contractors

  - If Lifetime Exposure is Greater Than Age in Years, Limit Exposure to:

    1991 - 1992 -- 2 Rem/Year From All Sites

  - Onsite Exposure Limit:

    1990 - 1992 -- 3 Rem/Year. Not to Exceed
    4.5 Rem/Year From All Sites

EXPOSURE REDUCTION INITIATIVES

- Short Term Initiatives (1987-1990) -- Completed
  - Increased ALARA Awareness
  - Work Practice Review
  - Construction Work Efficiency
  - Core Exit Thermocouples Grayloc Flanges
  - Spent Resin Processing Equipment
  - ALARA Installation Reviews
  - Cobalt Reduction Program

EXPOSURE REDUCTION INITIATIVES

- Long Term Initiatives (Beyond 1990)
  - Longer Fuel Cycles - Zircalloy Conversion
  - High Primary Side pH
  - Zinc Injection
  - Ultrafine Filters (Installed)
  - Full System Decon (W-EPRI)

RITS
RADIOLOGICAL INFORMATION TRACKING SYSTEM

- Third Generation In House Computerized Health Physics Records System Development Started in Mid 1989
  - Expected Online First Quarter 1991
  - Performs Basic Dosimetry Issue, Exposure Tracking, ALARA Reports Similar to Present Helpore System
  - Enhancements Over Present System:
    - Multiple Badging
    - Electronic RWP
    - Online Access Control
    - Survey Data Maps
    - Bioassay
    - Respirator Issue
    - Contamination Incident Data

HEALTH PHYSICS APPRAISALS

- Appraisal Program Covers the Following Areas Over a 3-Year Period:
  - Exposure Control
  - Surveillance
  - Radwaste
  - ALARA
  - Training
  - Facilities and Equipment
- 1990 Appraisal of the ALARA Area, Including Applicable Training, Facilities, and Equipment, was Completed
- In 1991 - Radwaste and Surveillance Appraisals
- In 1992 - Exposure Control Appraisals
- Staffing to Accomplish Appraisals Has Improved

PERSONNEL DOSIMETRY LABORATORY

- State-of-the-Art System in Service Since January 1, 1990
- NVLAP Accredited
- 3 Harshaw 8800 System Automatic Readers With Harshaw LIF Chip Dosimeter
- Enhancements
  - Single Dosimeter Provides Beta, Gamma, and Neutron Dose
  - Glow Curves
  - Hot Nitrogen Gas Heating
  - Monitoring of Photomultiplier Tube Voltage, Noise, QC Cards with Reader Shutdown if Tolerance Band Exceeded
  - Bar Codes
  - Dosimeter Case Designed by NU

OTHER EXAMPLES OF RADIOLOGICAL ENGINEERING SUPPORT ACTIVITIES

- ☐ Thermal Shield Waste Characterization and Shipping
- ☐ Fuel Clad Performance Tech Spec
- ☐ Radiological Environmental Monitoring Program
- ☐ Health Physics Standard Practices Manual
- ☐ Radwaste Burial Site Strategic Plan

CHEMISTRY

Cycle 16 Fuel Monitoring

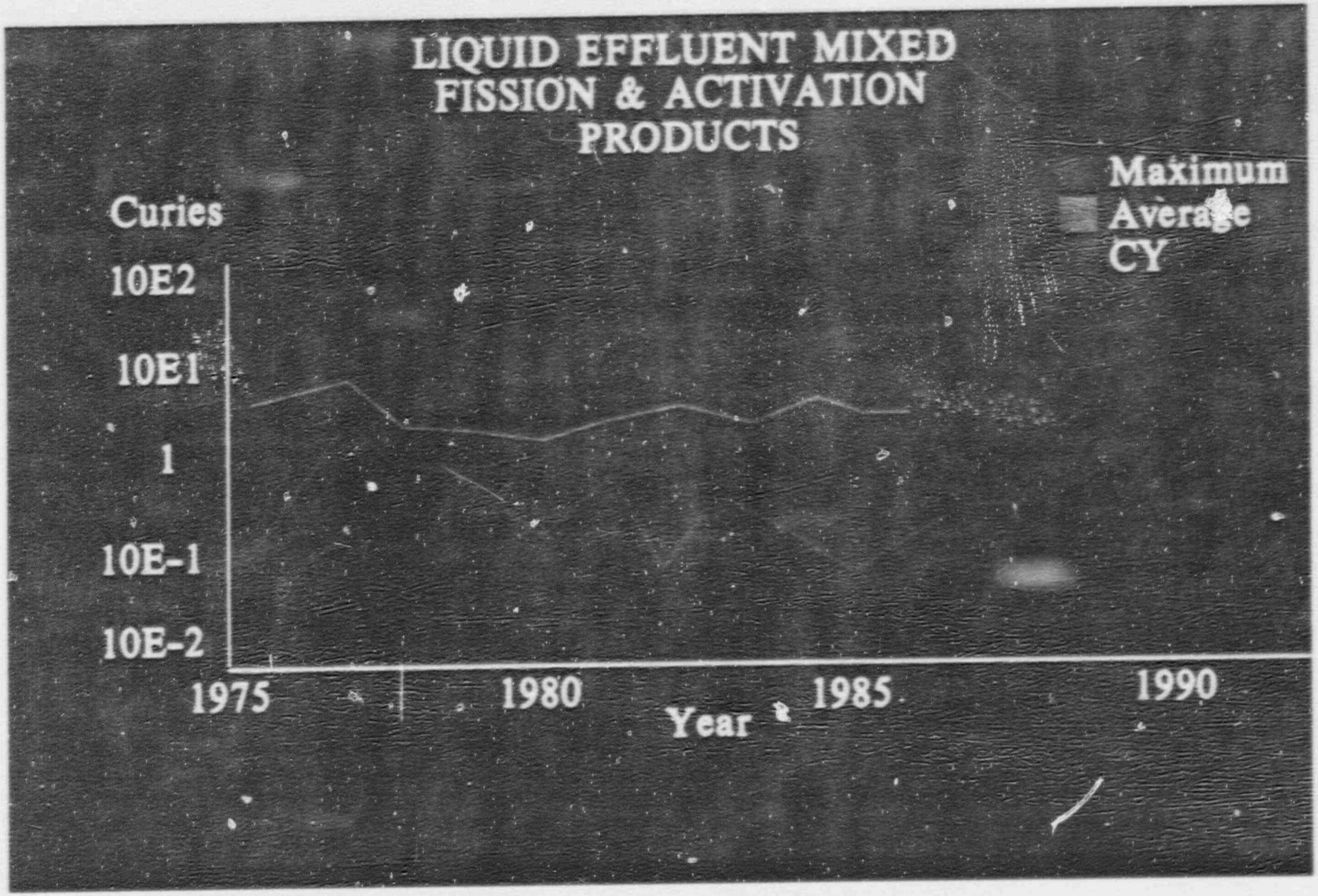
Radionuclide Effluents

AUGMENTED FUEL MONITORING PROGRAM

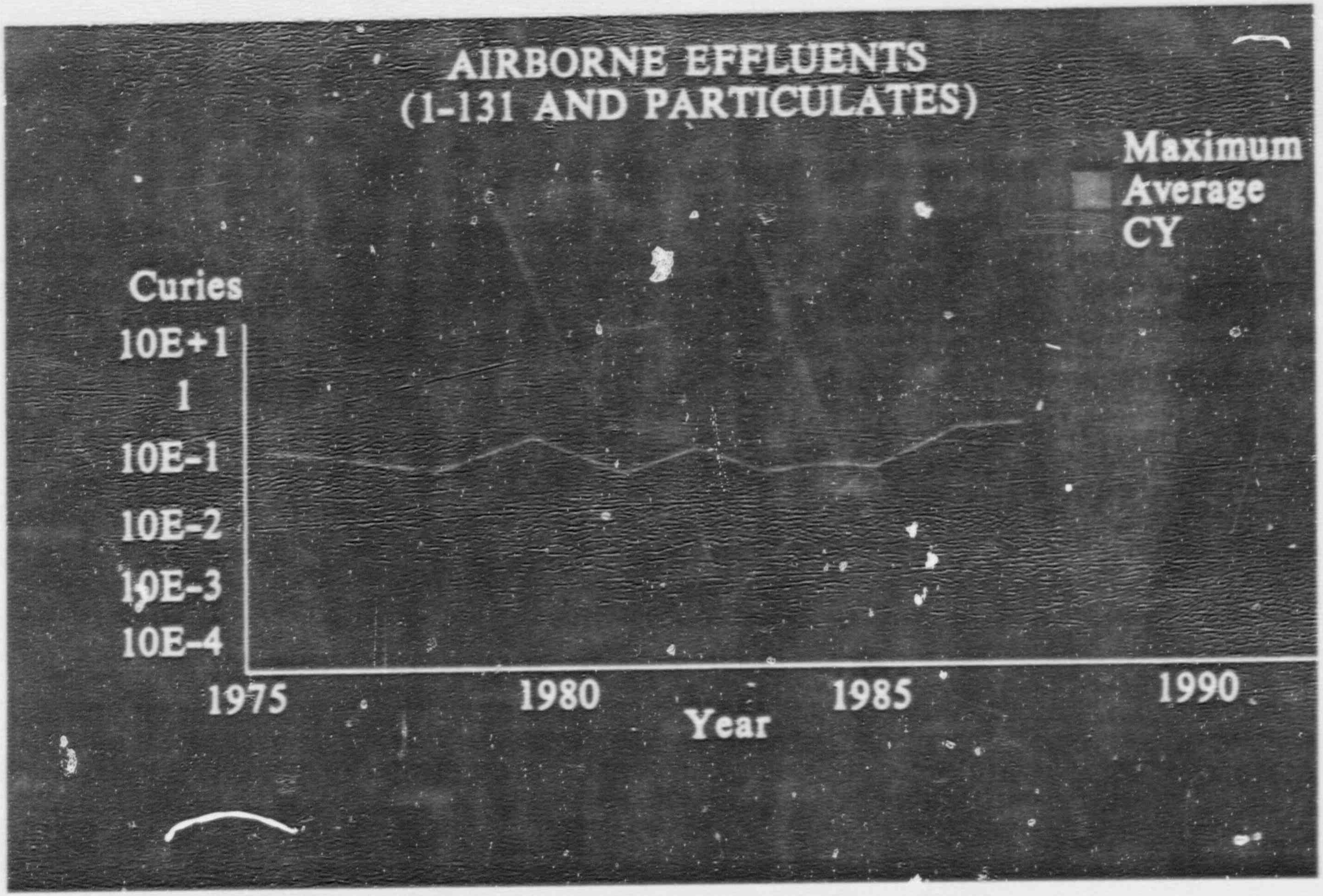
- ❑ Cycle 15 Fuel Damage
- ❑ Augmented Monitoring for Cycle 16
- ❑ Xenon Pin Equivalent (XPE) Calculation:
  - Developed In-House
  - XPE of 160 = 160 Cycle 15 Type Defects
- ❑ Technical Specification Change Approved January 4, 1991

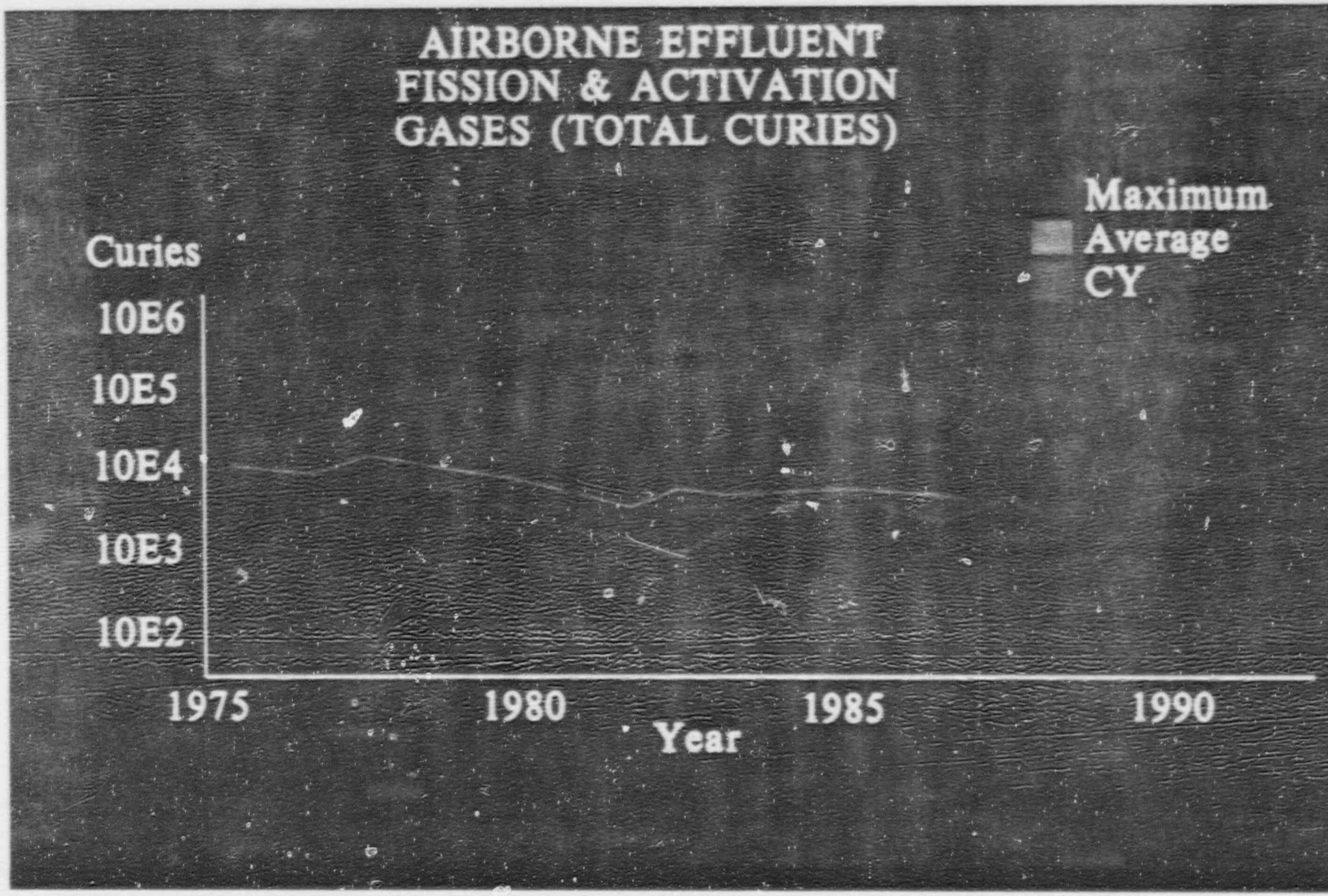
CYCLE 16 FUEL PERFORMANCE

- Indications of Approximately 20 Pins With Cycle 15 Type Defects
- Indications of 2 Pins With Conventional Failures
- Radiochemistry Analysis Indicates Second or Third Burn
- Evaluating 100% Core Offload, UT Inspection and Limited Reconstitution During Cycle 16 Outage



LIQUID EFFLUENT MIXED
FISSION & ACTIVATION
PRODUCTS
Curies
10E2
10E1
1
10E-1
10E-2
1975
1980
1985
1990
Year
Maximum
Average
CY


AIRBORNE EFFLUENTS
(1-131 AND PARTICULATES)
Maximum
Average
CY
Curies
10E+1
1
10E-1
10E-2
10E-3
10E-4
1975
1980
1985
1990
Year


AIRBORNE EFFLUENT
FISSION & ACTIVATION
GASES (TOTAL CURIES)
Maximum
Average
CY
Curies
10E6
10E5
10E4
10E3
10E2
1975
1980
1985
1990
Year

MAINTENANCE/SURVEILLANCE

Staffing

Maintenance Programs

Technical Specification Upgrade

Integrated Leak Rate Test

Steam Generator Status

Thermal Shield Removal

STAFFING

- Additions
  - Add One Maintenance Supervisor (3 New Supervisors)
  - Upgrading Selection Process for Maintenance Supervisors
  - 3 New Positions - 2 Technicians and 1 Engineer
- Improvements
  - Training
    - Job Task Analysis (All Duties Including Balance of Plant)
    - Second Round Accrediation
  - Procedures
    - Quality
    - Quantity
    - Balance of Plant

MAINTENANCE PROGRAMS

- ❑ Predictive Maintenance
- ❑ Check Valves
- ❑ Motor Operated Valves

TECHNICAL SPECIFICATON UPGRADE

- Revised Technical Specifications Have Been Implemented
- Development of Surveillance Database Complete and Validated
- Computer Tracking Capability
    - D[illegible]ate Tracked
    - Date Completed and Automatically Computes Next Due Date
- Procedure Revised and Updated
    - Trial Distribution of Reports In-Progress
    - Full Implementation Scheduled for 2nd Quarter 1991

CONTAINMENT INTEGRITY

- Integrated Leak Rate Test
  - In 1989 the "As Found" Test Passed
  - Program a Success as a Result of
    - Valve Replacements
    - Penetration Modifications
    - Aggressive Preventive/Corrective Maintenance
- Local Leak Rate Testing
  - Improved Test Instrumentation
  - Procedure Enhancements
    - Use of Detailed Graphics
    - Emphasis on Quality of Content
- Future
  - 1991 - Modify Penetrations to Eliminate Water Collection Methods
  - Thereafter - Philosophy

    Continual Reassessment of Reliability of Containment Isolation Methods

STEAM GENERATOR STATUS

- 1989 - 1990 Work Scope
  - 100% Inspection of SG Tubes
  - Plugged Identified Tubes with "Plug A Plug"
  - Plugged Additional Tubes Based on Probability for Leakage & Heat Numbers
  - Extensive Use of Robotics to Minimize Exposure

- 1991 Outage Plans
  - Inspection Plans Being Evaluated

THERMAL SHIELD REMOVAL

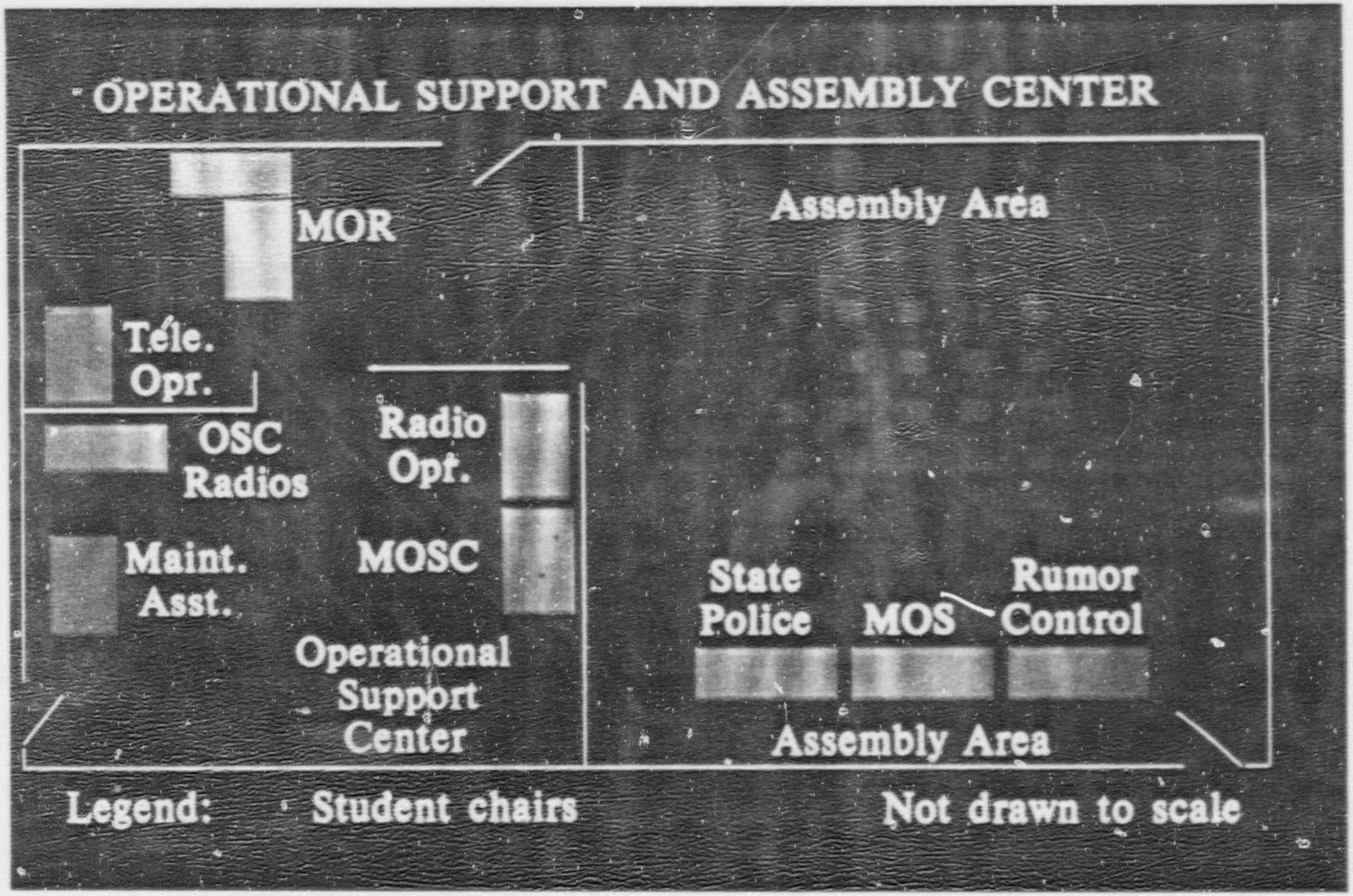
- 1987 - 1988 Refueling Outage
  - Repairs Not Fully Effective
- 1989 - 1990 Refueling Outage
  - Conservative Decision to Remove Rather Than Repair
    - Eliminate Potential for Future Extended Outages
    - Removal Utilized Method to Minimize Metal Particles
    - Extensive RCS Cleanup to Avert Further Fuel Damage

EMERGENCY PREPAREDNESS

- ☐ 1990 Annual Drill
- ☐ E-Plan Initiatives

1990 ANNUAL DRILL

- Strengths
- OSC Issue
  - OSC Staffing
  - Command and Control
  - OSC Facility


OPERATIONAL SUPPORT AND ASSEMBLY CENTER
MOR
Tele. Opr.
OSC Radios
Maint. Asst.
Radio Opr.
MOSC
Operational Support Center
Assembly Area
State Police
MOS
Rumor Control
Assembly Area
Legend: Student chairs
Not drawn to scale

INITIATIVES

- Drills
- Simulator Exercise in 1991
- New Emergency Notification and Response System
- ERDS Support

EMERGENCY PREPAREDNESS DRILLS 1990

- Monthly — *Operations and Security drill observed by a Station Emergency Preparedness Coordinator. These drills usually require event classification based on the EAL Table, redeployment of the Security Force into a response mode and fulfilling of other procedural requirements. These drills are 15 to 45 minutes in duration.*

- March 12, 16 & 19 — *Mini-drills involve two full sections of the Station Emergency Response Organization. Scenarios are developed from previous exercise/drill packages. The drills are usually 3 to 4 hours in duration.*

- April 8 — *Annual pre-exercise rehearsal drill involves two sections (Exercise team) of the Station and Corporate Emergency Response Organization. This drill scenario is usually the previous year annual exercise. The drill is usually 4 to 5 hours in duration.*

  Credit was given for the Health Physics portion of this drill.

EMERGENCY PREPAREDNESS DRILLS 1990

- September 5, 12, 19 & 26 — *Health Physics drill conducted with Chemistry Department. The drill was a post accident sampling team training drill conducted by the Nuclear Training Department.*

- September 17 — *A call-in drill was conducted for Station and Corporate Emergency Response Organization to respond to a radiopager message. There was no requirement to report to the site for this drill. The drill was about 1 hour in duration.*

- October 3 — *Annual Medical Drill. Involves the Station Medical Staff, Health Physics Department and the Security Department. The local ambulance and hospital staffs fully participate in the drill. The drill is usually 3 to 4 hours in duration.*

EMERGENCY PREPAREDNESS DRILLS 1990

- October 24 — *Annual Fire Drill. This drill involved both the Haddam Neck Fire Volunteer Department and the East Hampton Volunteer Fire Department. The training and drill was conducted by the Nuclear Training Department.*

- November 20 — *Simulator based drill. This drill verified the following:*
  - *physical changes to OSC*
  - *reorganization of OSC*
  - *personnel change*
    - *added technical assistant to OSC Manager*
  - *verified use of simulator for drill*

Note: There were also two unannounced call-in drills to test and verify the Emergency Notification and Response System (ERNS)

SIMULATOR USAGE EVALUATION

- Anticipated Usage in May 1991, Partial Participation Exercise
- Project Initiated in June 1990
- Shakedown Test November 20, 1990
- Additional Tests Scheduled for April 1991
- Evaluation Report Planned for June 1991

ENRS
EMERGENCY NOTIFICATION AND RESPONSE SYSTEM

- ❑ Online October 22, 1990
- ❑ Upgrades Present Callback Recorder System Used to Provide State and Local Officials with Detailed Incident Reports
- ❑ Instant Report on Call-In Status
- ❑ Expedites Control Room Notification Duties
- ❑ Computerized Digitized Voice System
- ❑ Separate Phone Line for Each State and Local Organization
- ❑ Also Used for Site and Berlin EOC Response Organization Call-Ins

ERDS

- Letter of Agreement to Participate Sent August 14, 1990
- Project to Implement ERDS will be in 1991
- Plan is to Tie into Existing NU OFIS Database now Supporting TSC, EOC

SECURITY

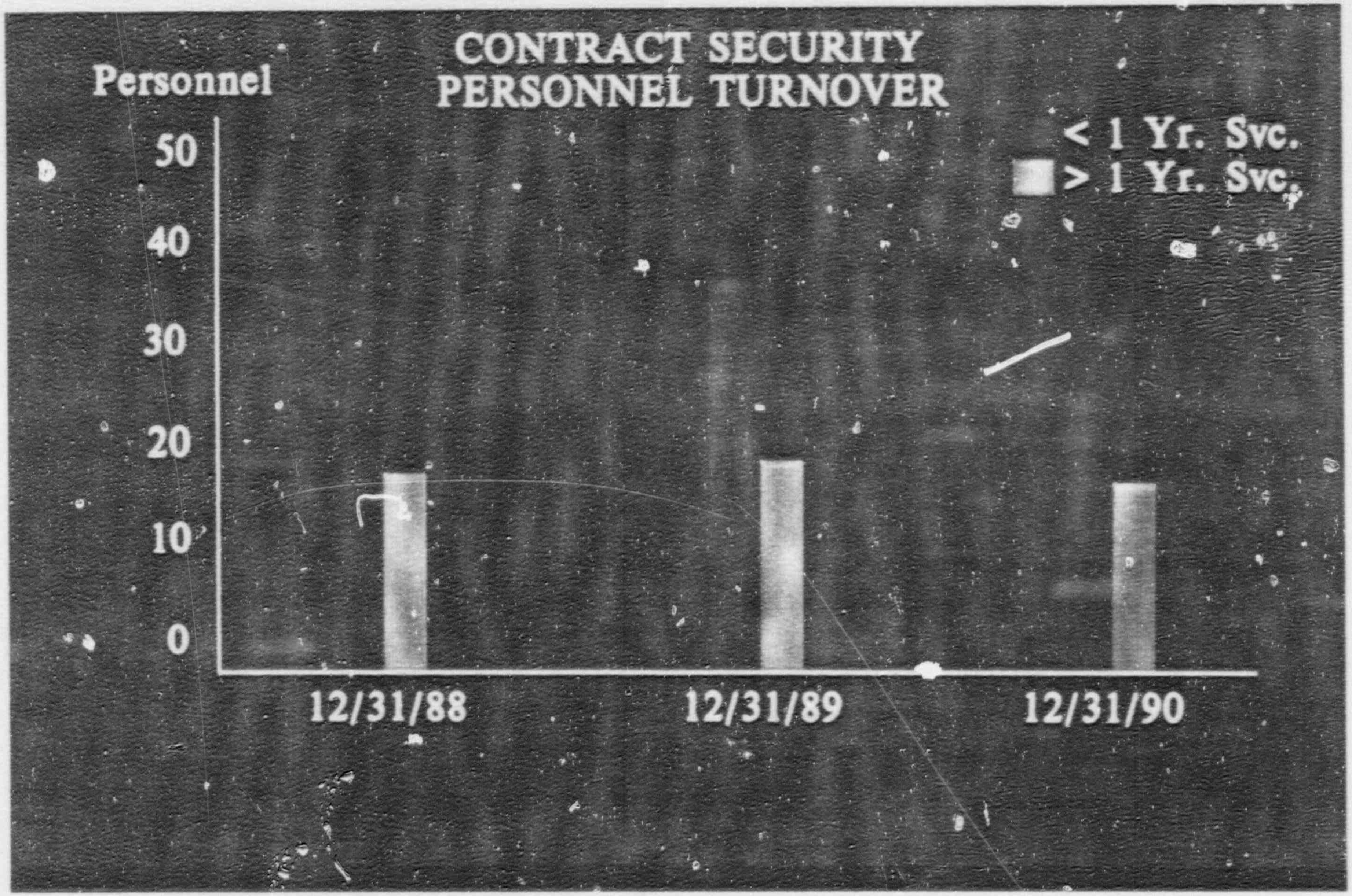
Guard Force Retention

Initiatives

Fitness for Duty Program

GUARD FORCE RETENTION

- Attitudinal Survey
- Retention Initiatives
  - 401K Plan
  - Weapon Qualification Competition
  - Service Recognition
  - Enhanced Guard Force Training
  - Station/Contractor Communications
- Results
  - Turnover Remains Higher Than Desired
  - Indications of Positive Attitude Changes


CONTRACT SECURITY
PERSONNEL TURNOVER
Personnel
50
40
30
20
10
0
< 1 Yr. Svc.
> 1 Yr. Svc.
12/31/88
12/31/89
12/31/90

SECURITY INITIATIVES

- ❑ Upgrade Intrusion Detection System
- ❑ New Computer System
- ❑ Training Enhancements
- ❑ Vital Area Upgrades
- ❑ Security/Operations Drills
- ❑ Contracted Security Force Contingency Drills
- ❑ Self Assessments

FITNESS FOR DUTY PROGRAM

- NU Committed to Drug-Free Environment
- NRC Audit Results/Corrective Actions

ENGINEERING/TECHNICAL SUPPORT

Major Tasks Completed

Current Efforts

Service Water System

Auxiliary Feedwater System

Active Project Assignment Comparison

MAJOR TASKS COMPLETED

- Thermal Shield Removal
- Reactor Protection System Modernization
- Nuclear Instrumentation System Replacement
- New Switchgear Building
- Revised Technical Specifications
- Fuel Reconstitution

CURRENT EFFORTS

- Auxiliary Feedwater System
- Service Water Temperature/Flow Concerns
- MOV - Generic Letter 89-10
- Check Valve Program
- Electrical Distribution System Functional Inspections

SERVICE WATER SYSTEM

- ❑ Biofouling Control Program
- ❑ Safety Related Service Water Heat Exchangers Test Program
- ❑ Inspection and Maintenance of Piping and Components
- ❑ Design Basis Reconfirmation
- ❑ Maintenance Practices, Operating and Emergency Procedure Training Confirmation

AUXILIARY FEEDWATER SYSTEM

- ❑ Change Administrative Controls DWST Minimum from 50K to 70K Gallons
- ❑ Improve Procedure for AFW Alternate Supply Path
- ❑ AFW Dependence on Control Air Identified and Resolved in Short Term
- ❑ AFW Dependence on Manual Operator Action to Fully Open Terry Turbine Steam Admission Valve Identified and Resolved in Short Term

AUXILIARY FEEDWATER SYSTEM

- Modifications to Accomplish
  - Turbine Trip/Overspeed - Loss of Control Air
  - AFW System to Automatically Provide Design Flow
  - Correct Human Engineering Discrepancies
  - Correct Design Basis Discrepancies
- Modifications Being Considered
  - Microprocessor Governor System
  - New Check Valve
  - Valve Modifications

ENGINEERING/TECHNICAL SUPPORT

ACTIVE PROJECT ASSIGNMENT COMPARISON

	1990	1991/1992
Millstone 1	21	13
Millstone 2	16	16
Millstone 3	23	21
Conn Yankee	27	22
Total	87	72

Only Includes PA's
Resulting in Construction

ENGINEERING/TECHNICAL SUPPORT

Modifications

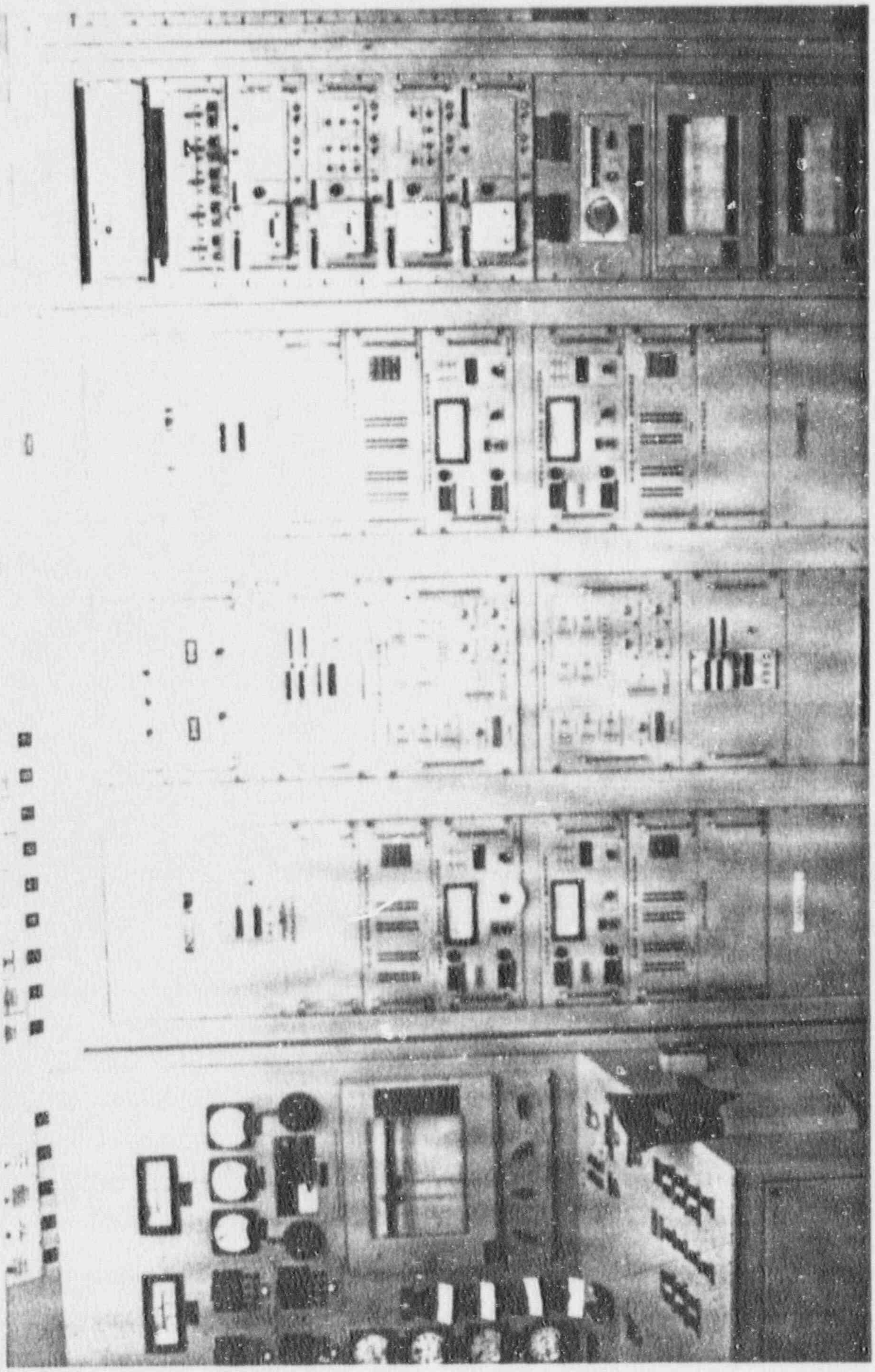
Preoperational Testing 1989 Outage

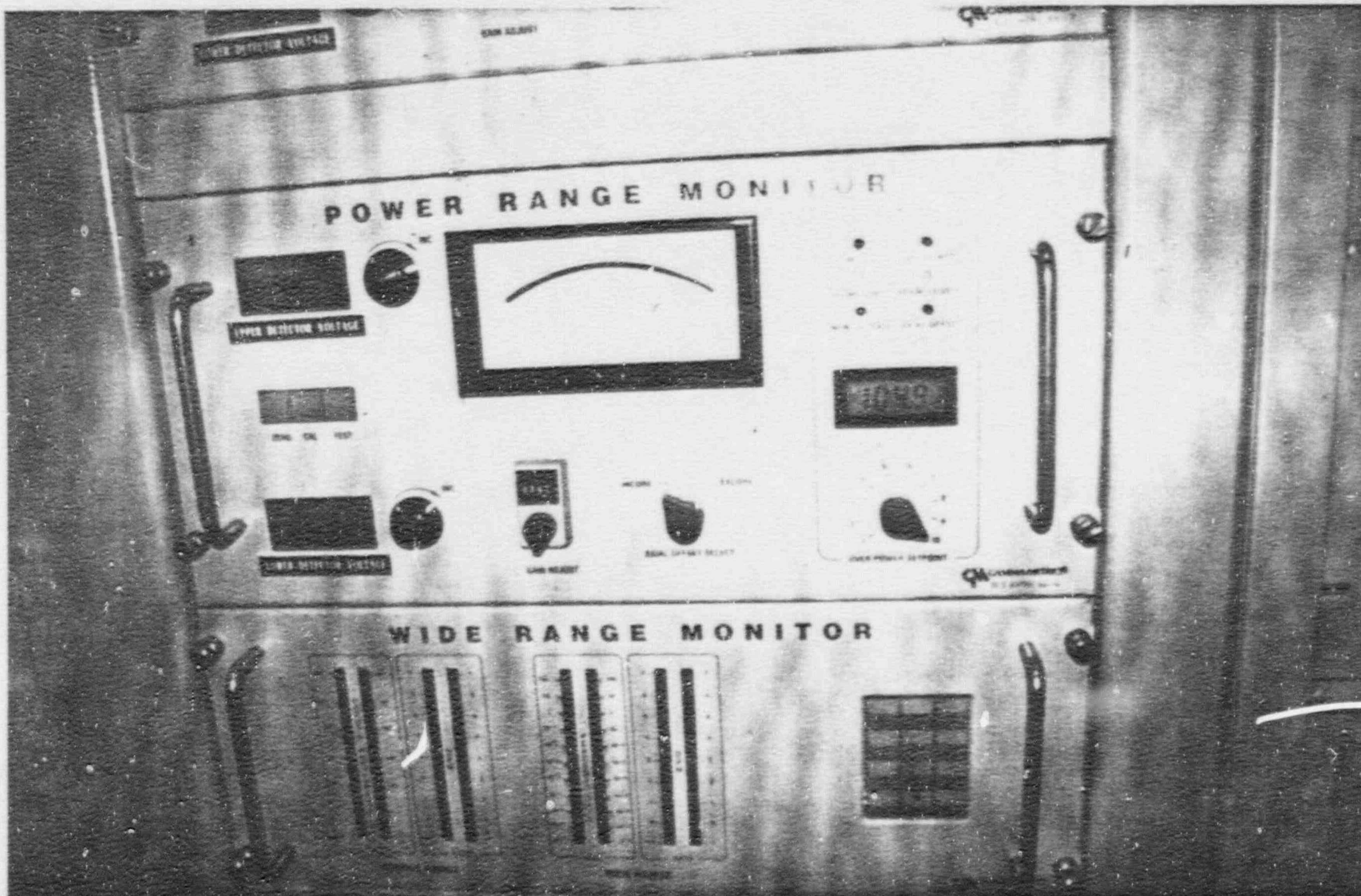
Procurement

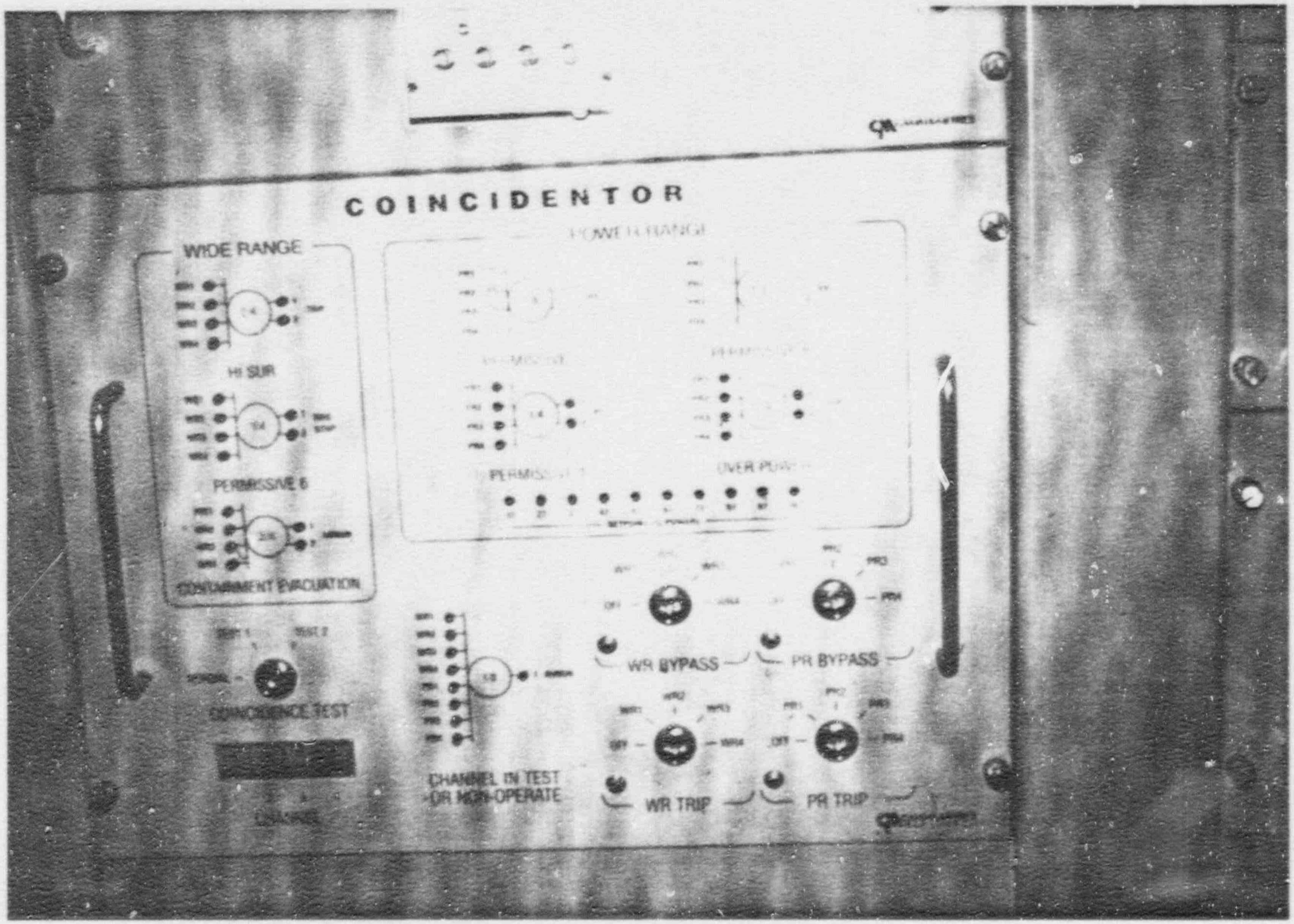
Staffing

MODIFICATIONS

- Philosophy
  - Replace Obsolete/Problem Equipment
- Integrated Safety Assessment Program
- Planning/Scheduling
- Installation
- Testing
  - Preoperational Testing - 1989
    - Dedicated Test Coordinator
    - Demonstrated Safeguards Operability
    - Power Supply Independence
  - Procedure - General




POWER RANGE MONITOR
WIDE RANGE MONITOR


COINCIDENTOR
WIDE RANGE
HI SUR
PERMISSIVE 6
CONTAINMENT EVACUATION
COINCIDENCE TEST
CHANNEL IN TEST
OR NON-OPERATE
WR BYPASS
PR BYPASS
WR TRIP
PR TRIP
OFF

MODIFICATIONS

- Cycle 15 (1989 - 1990) Refueling Outage Modifications
  - Reactor Protection System Modernization
  - Appendix R Switchgear
  - NIS Replacement
  - Thermal Shield Removal
  - CAR Fan Repowering
  - Penetration Seal Survey
  - Fire Protection Upgrade
  - ECCS Modifications

- Cycle 16 (1991) Operating and Refueling Outage Modifications
  - Flux Mapping System Replacement *
  - Plant Security Computer Replacement
  - ALARA Modifications
  - Hypochlorite System Upgrade
  - Appendix J Modifications *
  - Auxiliary Feedwater Modifications *
  - Mid Loop Operation Instrumentation *
  - Charging Flow Mini Flow Valves *

* Outage

MODIFICATIONS

- Cycle 17 (1992 - 1993) Refueling Outage Modifications
  - Modernize Feedwater Control
  - Seismic Qualification of Safety Piping
  - Relocation of DG Control
  - Pressurizer Spray System
  - Relocate Control MOVs 1, 2, 3, & 4
  - Replace 345KV Supervisory Control Equipment
- Future Modifications
  - HP Turbine Replacement
  - Spent Fuel Pool Re-Rack

PROCUREMENT

- Staffing
  - Established March 1989
  - Reports to Engineering Manager
  - Current Staffing - Six

    (1 Supervisor; 4 Engineers; 1 Technician)

- Recent Activities 1990
  - Training Completed - EPRI

    CGI - Commercial Grade Items

    TERI - Technical Evaluation of Replacement Items
  - Completed Commercial Grade Item Sample - 3200 Man Hours
  - Corporate Procedures Revised 1990
  - Completed Audit of Procurement Activities 10/90

STAFFING

- Adding 3 Engineers
  - 2 Procurement
  - 1 ISI
- Quality
  - Average Years of Experience
    - Management 13 Years
    - Staff 10 Years
  - Training
    - Engineering Representation on Training Program Control Committee

SAFETY ASSESSMENT
QUALITY VERIFICATION

JOHN P. STETZ

STATION DIRECTOR

SAFETY ASSESSMENT/ QUALITY VERIFICATION

- ❑ Integrated Safety Assessment Program (ISAP)
- ❑ Safety Initiatives
- ❑ Self Assessments
- ❑ Procedure Upgrades
- ❑ Safety Ethic
- ❑ Training
- ❑ Licensing Submittals
- ❑ Site Visits
- ❑ Nuclear Review Board

INTEGRATED SAFETY ASSESSMENT PROGRAM (ISAP)

- Benefits
  - Safety Improved in most Efficient Manner
  - "Integrated" Review of Plant Specific Issues
  - Plant Specific Review and Evaluation of "Generic" Requirements
  - Effective Regulatory/Utility Interface
  - Improved NRC and NU Resource Management

- Lessons Learned
  - Provides Rational Basis for Project Implementation Decisions
  - Provides Unit Director Meaningful Decision Making Input
  - Provides Opportunity to Enhance Regulatory/Utility Communication
  - Appears to be Very Complimentary to IPE Process

- Conclusions
  - Connecticut Yankee Believes that ISAP:
    - Has Improved Public Safety
    - Provides Logical, "Integrated", Programmatic Resource Allocation
    - Improves Internal and External Communication
    - Provides Mutual Benefit to NRC and NU Management
    - "Is the Decision Making Tool for the 1990's"

SAFETY INITIATIVES

- Continued Commitment to Plant Safety
- Living PRA
  - High Relative Ranking and Scheduling Assigned to Modifications That Have a High Public Safety Benefit
- Self Identification of Potential Problems
  - Low Reporting Threshold
  - Prompt Corrective Action
  - Root Cause Determinations

- Core Melt Frequency Reduction Task Force
  - Task Force Multi-Disciplined - 12 Individuals
  - Goal < 10 E-4 CMF
  - Current CMF Estimate 4 X 10 E-4/Year
    - Reduction of 2 Since 1987-1989 Time Frame
    - Reduction of 4 Since 1986 Time Frame

SELF ASSESSMENT

- Human Performance Enhancement System (HPES)
  - 5 HPES Evaluations Performed in 1990 by Nuclear Safety Engineering (NSE)
- Independent/Support of Root Cause Evaluations
  - 3 Root Cause Evaluations Performed 1990
- Operating Experience Reviews
  - Routine Reviews of:
    - Station Licensee Event Reports
    - Station Plant Information Reports
    - INPO SOER's

PROCEDURE UPGRADES

- Use of Graphics
  - Instrumentation and Control
  - Mechanical Maintenance
  - Electrical Maintenance
- Status
  - 2 Year Procedure Upgrade Complete
- Has Assisted in Improved Procedure Compliance as Described in SALP Report

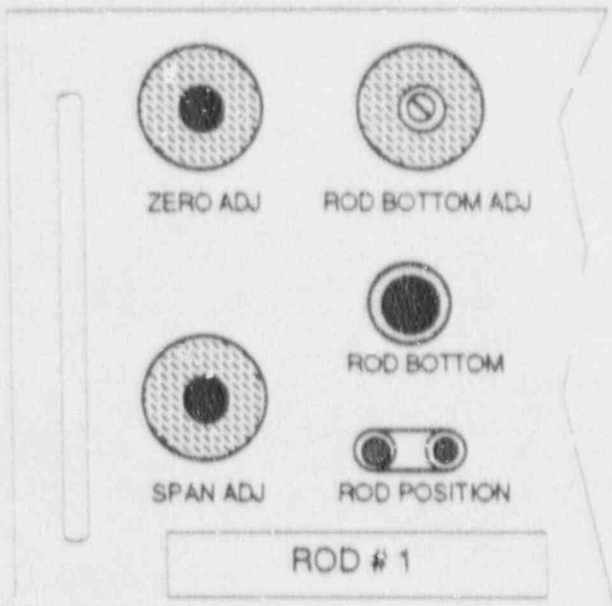
6.2.4 IF As Found zero step readings are acceptable, proceed to step 6.2.5; OTHERWISE proceed as follows to adjust RPI drawer ZERO potentiometer(s) (Figure 6.2-1):

> **NOTE**
>
> Care must be used when making the ZERO adjustment since the voltage will not reverse polarity and over-adjustment will result in a deadband at rod bottom.

a. SELECT rod using MCB selector switch.

b. TURN ZERO potentiometer CCW while observing DVM, until voltage begins to rise.

c. TURN ZERO potentiometer slowly CW while observing DVM, until zero or minimum voltage is obtained (see note above).



RPI Drawer Front Panel

Figure 6.2-1

6.2.5 RECORD DVM As Left zero step readings in Table 6.2-1.

6.2.6 HAVE Operations withdraw Bank C until its odometer indicates 320 steps.

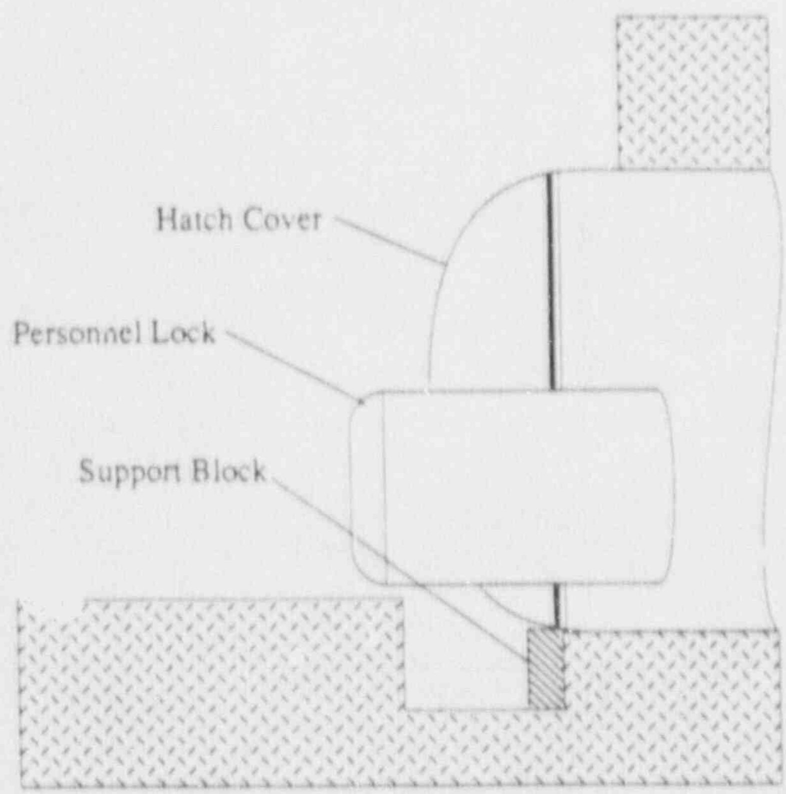
6.2.7 VERIFY Bank C odometer indicates 320 steps and RECORD in Table 6.2-1.

CAUTION

The hatch dome must remain supported by the yard crane until the eighteen (18) bolts are tightened and support block is positioned under cover.

a. INSTALL a support block under the hatch cover, as shown in Figure 1, to support the dead weight of the cover.

Figure 1
Support Blocks for Hatch Cover



b. PLACE eighteen (18) swing bolts to the lock position and SNUG the bolts per the sequence of Figure 2.

c. TIGHTEN the bolts per the sequence of Figure 2 using as many passes as required to have metal-to-metal contact between the cover and the containment flange.

d. REMOVE the lift gear from the hatch.

NOTE
Perform this section only if inspection of drive sleeve assembly (section 6.21) warrants further disassembly.

6.22 Worm and Belleville Spring Pack Disassembly and Inspection (Refer to Figure 13)

_____ 6.22.1 REMOVE and DISASSEMBLE wc.. rom belleville spring pack assembly per the following.

a. SEPARATE worm from worm shaft bearing cartridge by removing retaining ring.

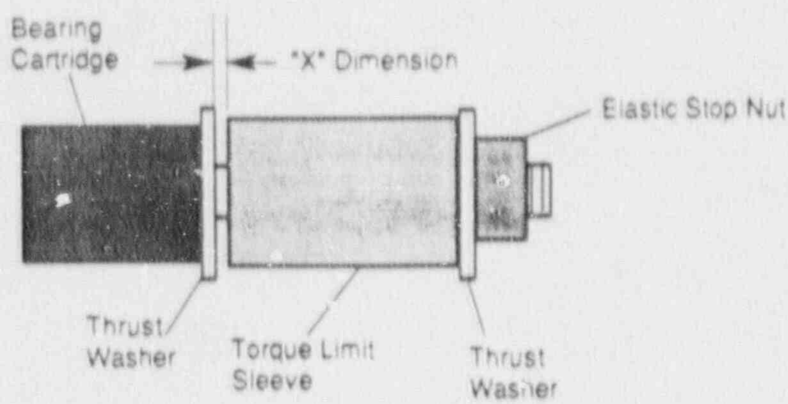
b. REMOVE worm bearing nut and bearing, if necessary.

6.22.2 DISASSEMBLE belleville disc assembly per the following.

_____ a. MEASURE dimension "X" before removal of elastic stop nut (See Figure 14). With the torque limit sleeve tight against the elastic stopnut end thrust washer measure dimension "X" in at least two places and RECORD measurements below.

Dimension "Y" measurements ______ ______ Inches

Figure 14
Spring Pack "X" Dimet sion



SAFETY ETHIC

- NEO's Safety Ethic Principles
  - Safety Ethic Principles Issued to all NE&O Personnel November 1990

- Enhanced Nuclear Safety Concerns Program
  - Employee Awareness Campaign Conducted on NSCP Enhancements November/December 1989
  - NSCP Office Opened January 1990
  - NSCP Director Conducted Formal Orientation Program, and Emphasized 4 Options Available and Employee Protection

    <u>Options</u>

    – Supervision and Chain of Command
    – Nuclear Safety Concern Program
    – Nuclear Review Team
    – NRC

    <u>Protection</u>

    – Employees are Protected from Intimidation/Harassment per NU Policy Statement

- NRC Team Inspection October 1990
  - Majority of Employees Comfortable with Program
  - NSCP Evaluating NRC Team Inspection and Internal Feedback to Make Further Enhancements

NE&O'S SAFETY ETHIC PRINCIPLES

The mission of Nuclear Engineering & Operations (NE&O) is safe, environmentally sound, dependable, and economic generation of electrical energy utilizing nuclear power. The foundation of this mission is the commitment of each individual in NE&O to a Safety Ethic which creates a pro-active and positive Safety Culture. The Safety Ethic is embodied in the following principles:

* The key resource of NE&O is its people and their professionalism.

* NE&O Personnel must:

  — Create a state of mind which emphasizes safety and reliability through conservative decision making and recognizes that a high capacity factor is a natural consequence;

  — Recognize that the potential risks inherent in nuclear technology necessitate that safety considerations pervade all activities;

  — Accept personal and moral responsibility to ensure the safety of the public, utility and contractor personnel in our nuclear activities;

  — Maintain a state of mind which promotes the use of: (1) effective root-cause assessment of events and deficiencies, (2) timely and effective corrective actions, (3) "what if" questions, (4) attention to detail, and (5) rigorous and constructive self-criticism;

  — Perform work in accordance with applicable procedures and requirements.

* NE&O Management must:

  — Establish and implement policies and procedures which ensure correct safety practices;

  — Create an organizational structure with clearly communicated lines of responsibility and high accountability;

  — Select, train and fill positions with the best-qualified and effective individuals who embody the NE&O Safety Ethic;

  — Practice open communication characterized by active listening and welcome the identification of potential safety issues;

  — Establish oversight review processes for nuclear activities.

E J Mroczka

E. J. Mroczka
Senior Vice President

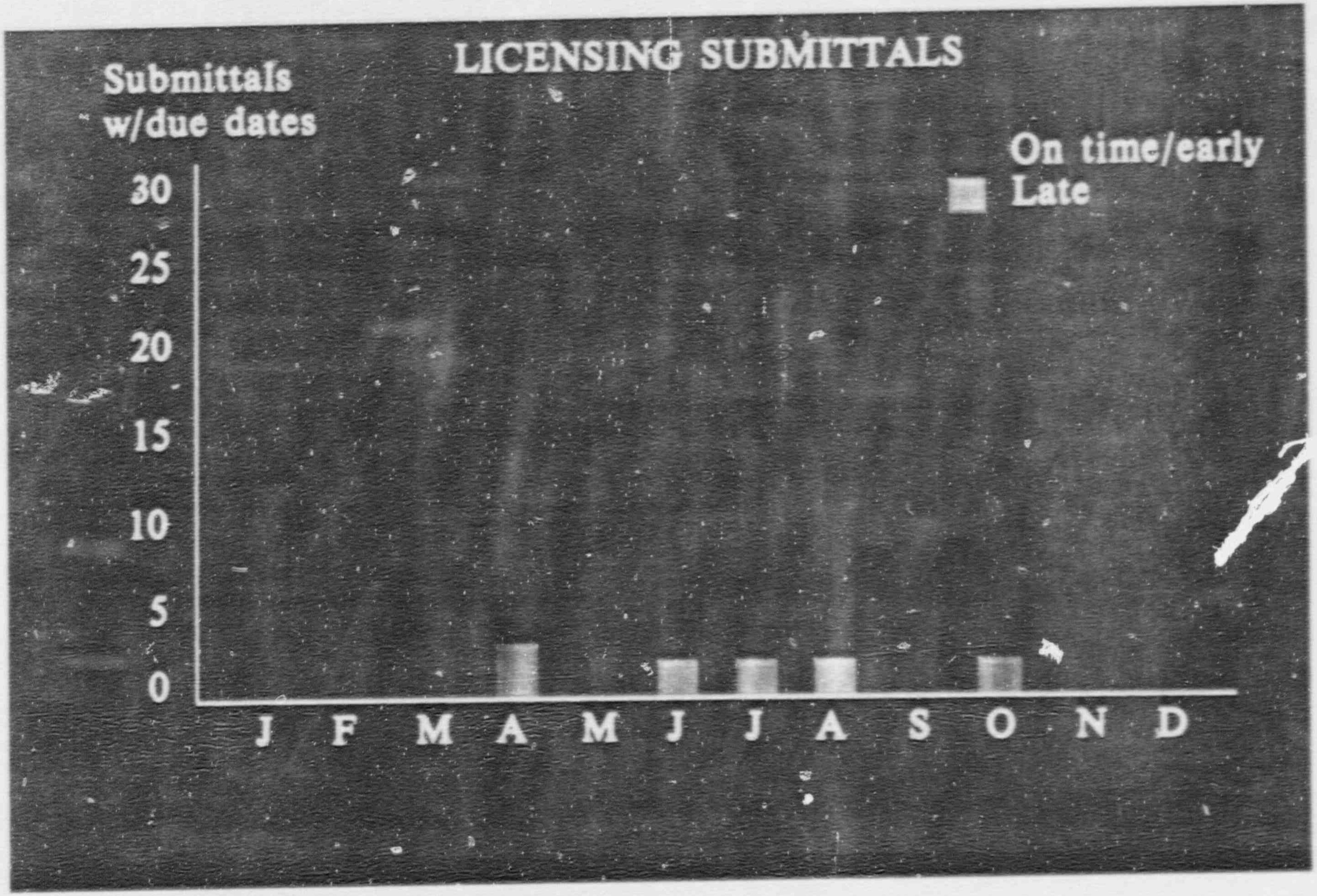
11-28-90

TRAINING

- Engineering Certification
- 2nd Round Technical and Operator Training Accredited

LICENSING SUBMITTALS

- ❑ Track Timeliness by Monthly Report
- ❑ Zircalloy Conversion Progressing Well
- ❑ Periodic Meetings Between NU/NRC


LICENSING SUBMITTALS
Submittals w/due dates
30
25
20
15
10
5
0
On time/early
Late
J F M A M J J A S O N D

SITE VISITS

- NRC Commissioner Curtiss
- Russian Delegation
  - Feedback
    - Impressed with Procedures and Procedure Usage
    - Agreed to Develop Similar Symptom Based EOP's at Novovonezhskaya (Units 3 & 4) Simulator
    - Agreed to an Action Plan
    - Action Plan has been Approved by the Executive Steering Committee

NUCLEAR REVIEW BOARD

- In-House Multi-Discipline Group-Supervisor Level
- 19 Years Average Experience Per Person
- Reviews All Nuclear Safety Associated Documents
- Meets Monthly
- Monthly Walk-Throughs of Plant
- Pursue Issues Proactively
- Provided SR VP NEO with independent assessment of thermal shield debris removal program, STS conversion, switchgear building and RPS-NIS upgrade
- Provide SR VP NEO with annual evaluation of facility performance
- Annual audits of plant operations, maintenance, and training

SERVICE WATER SYSTEM SSFI

- Purpose of SSFI
  - Will the Service Water System Meet the Design Objective Established for the System?

- System Selection
  - Initial Selection Criteria
    - Importance to Safety
    - Complexity of Design
    - Accessibility
    - Number of Modifications
  - Initial Selectic
    - Residual Heat Removal
    - Service Water

- Basis for Final Selection
  - The recent design re-evaluation effort for the Service Water System provides an opportunity to evaluate the current NU design and modification process.

- Performance Objectives
  - Establish in Work Plan
    - Engineering/Design

      ED - 1 Design Basis

      ED - 2 Modification Control

      ED - 3 As-Built Design

      ED - 4 Configuration Control
    - Maintenance

      SM - 1 Preventive and Corrective Maintenance

      SM - 2 Mainteance Training

- Surveillance/Testing

  ST - 1 Testing

  ST - 2 Surveillances

  ST - 3 Training

- Operations

  SO - 1 Component Installation

  SO - 2 System Line-Ups

  SO - 3 Procedures

  SO - 4 Operational Experience

  SO - 5 Operator Training

- SSFI Team Organized to Review Objectives in Each Functional Area

– Engineering/Design	3 Engineers 1 Analyst
– Maintenance	2 Engineers
– Surveillance/Testing	2 Engineers
– Operations	1 Shift Supervisor

❑ Observations

- Potential Failure to Meet Performance Objective

<u>Status of Observations</u>

	Total	Resolved
Category A *A failure to meet a Performance Objective that has potential immediate safety impact or is potentially reportable and needs immediate attention by the unit.*	14	14
Category B *A failure to meet a Performance Objective that is important to unit reliability and safety but does not present immediate safety or reportability concerns.*	22	22
Category C *A failure to meet a Performance Objective that does not present safety, reliability, or reportability concerns.*	32	27

System Functionality

- Addresses Basic Purpose of the SSFI
- The Service Water System can Perform its Safety-Related Functions
- Issues that had the Potential to Jeopardize the Functionality of the System have been Resolved.

Overview of Functional Concerns

Safety Function	E/D	Maint.	S/T	Ops
Overall System Performance	4	1		
EDG Cooling	1			
RHR				
CTMNT Cooling	2			
SFP Cooling				
Isolation of Non-Safety Loops	1		1	

- Functional Concerns

  - Overall System Performance

    The thermal hydraulic analysis (THA) used to evaluate SWS performance was not adequately verified to actual plant conditions. Verification of the model used in the analysis was planned and took place during 1989 - 1990 refuel outage.

    The original design basis for the SWS required that the piping to safety related components be seismically qualified but that the piping downstream of these components need not be. However, the non-seismic piping is credited in the current THA. Engineering evaluated this concern and determined that down-stream piping could be folded into the SEP topic addressing the upgrading of seismically qualified piping and that the JCO for the SEP scope piping could be applied to the down stream piping.

    A significant amount of corrosion material/scale was found on inside of the SWS piping during the verification testing for the THA. SSFI Team was concerned about the impact of this material coming loose in the system. An evaluation by Nuclear Material and Chemistry determined that this scale was tightly adhered and would not jeopardize the system.

- Overall System Performance (cont.)

  The SEP review identified an interaction between Charging Lines and SWS inside containmnent. However, there was not enough energy in the charging line to break SWS lines. This concern was identified as a result of a misunderstanding by the SSFI Team of information presented in the SEP HELB report.

  The SWS pump discharge strainer cleaning program was a time based program rather than a performance based program. The SSFI Team recommended that cleaning be performance based (i.e., based on strainer dP). The program will be modified to do this.

- EDG Cooling

  A potential common mode failure of the EDGs was identified as a result of failure of the non-seismic SWS piping downstream of the EDG coolers and within EDG cubicles. This is a specific case of the "downstream piping" issue identified above. A review of the seismic analysis indicated that it actually encompassed the entire length of pipe within the EDG cubicle eliminating the concern.

– Containment Cooling

The Adams filter backwash system was not seismically qualified which potentially jeopardized the ability to provide containment cooling flow. This concern had already been identified independently when the SSFI Team reviewed this portion of the system. This Observation was initiated to identify specific concerns that should be addressed in the design change planned to resolve this issue.

The HELB piping interaction identified above could also have affected containment cooling.

– Isolation on Non-Safety Related Components

A heating steam line break could result in a harsh environment in the area of two SWS motor operated valves used to isolate the non-safety related portions of the system. These valves are not environmetally qualified. Failure to isolate these portions of the system could have had an adverse impact on the system flow balance. An evaluation by Engineering determined that a break of the heating steam line would not cause a transient requiring closure of these valves. Therefore the lack of environmetal qualification was acceptable.

- Isolation on Non-Safety Related Components (cont.)

The SWS isolation valves for the non-safety related portions of the system must close when an undervoltage (UV) condition is detected on specific electrical busses. Each of these busses have both A and B train detection circuits. These detection circuits were not being tested on periodic basis nor were A and B circuits being tested independently. The pre-operational test for the electrical system modification was modified to include this testing. A periodic test of these circuits will also be developed.

- ❑ General Comments
  - Design Basis Documentation Needs Improvement