U.S. NUCLEAR REGULATORY COMMISSION
REGION I

REPORT NUMBER:	50-309/96-09
DOCKET NUMBER:	50-309
LICENSEE NUMBER:	DRP-36
LICENSEE:	Maine Yankee Atomic Power Company 329 Bath Road Brunswick, Maine 04011
FACILITY:	Maine Yankee Atomic Power Station
INSPECTION DATES:	August 4, 1996 through September 14, 1996
INSPECTORS:	J. Yerokun, Senior Resident Inspector W. Olsen, Resident Inspector R. Ragland, Radiation Specialist
APPROVED BY:	Richard J. Conte, Chief Projects Branch 5 Division of Reactor Projects

EXECUTIVE SUMMARY

Maine Yankee Atomic Power Station
NRC Inspection Report 50-309/96-09

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a 6-week period of resident inspection; in addition, it includes the results of announced inspections by a regional radiation specialist and a regional projects inspector.

Operations

Operators demonstrated good technical knowledge and safety perspective during plant operations and they responded well to plant anomalies such as dropped CEA rods during reactor start up. Operators took good actions to ensure continued safe operation of the plant during severe weather. The approved procedure (AOP 2-40, Severe Weather) in place to address the situation was weak because it lacked specificity and did not address specific actions to be taken to deal with issues such as specific wind speeds, staffing requirements to deal with the expected severe weather, and specific power level restrictions.

Safety related systems, such as the Primary Component Cooling Water System, were maintained operable such that they would be capable of performing their design functions if required.

Maintenance

Maintenance and surveillance activities were conducted safely and in accordance with approved documents. Technicians and operators showed good knowledge of the work activities being conducted. Supervisors properly monitored work activities.

Activities such as the troubleshooting of the failed Reactor Protection System channel, and the surveillance testing of the Steam Generator Blowdown Isolation Valves were conducted well. However some instances were noted that require improvement, such as with the welding of RH-4 when a substitute weld wire was used (later found to be acceptable).

An apparent violation of NRC requirements was observed with the lack of calibration of the Emergency Diesel Generator room fan thermostats, although they were functionally tested. This failure to maintain proper calibration of equipment important to safety was contrary to 10 CFR 50 Appendix B, Criterion XI, Test Control, which requires that a test program to assure that all testing required to demonstrate structure, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures (see also Engineering section).

Engineering

Plant Engineering Department personnel provided good support to the plant at resolving emerging issues, such as developing and conducting the tests of the SCC and PCC valves and initiating the temporary modification for the Containment Spray Building (CSB) fan dampers. They showed good technical capability at developing and conducting the tests. The inspector noted that this reflected good support to operations from being provided by the engineering organization.

However, two apparent violations of NRC requirements were noted. One violation involved past operation with the potential for the CSB fan dampers to fail into a position that would have prevented the fulfillment of the safety function of the LPSI and CS systems. This was contrary to 10 CFR 50, Appendix B, Criterion III, Design Control, which requires in part that design control measures shall provide for verifying or checking the adequacy of designs. In another instance (see also Maintenance Section), the functionality of the PCC and SCC flow control valves had not been properly demonstrated. This appeared to be contrary to 10 CFR 50, Appendix B, Criterion XI, Test Control.

Plant Support

Maine Yankee continued to maintain an adequate radiological controls program. However, as a result of your discovery of an unplanned exposure that apparently occurred during an April 16, 1996, radwaste filter change out, an unresolved item was opened, pending the completion of your investigation, and additional NRC review of licensee survey and monitoring practices. (URI 50-309/96-09-01). (Section R1.1)

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Report Details

Summary of Plant Status

Maine Yankee began this inspection period in the cold shutdown mode. The plant had been shut down earlier to repair the Containment Air Recirculation Fans' Primary Component Cooling (PCC) Water pipes. Following the repairs, the plant was made critical on August 14, 1996. However, due to problems discovered with the High Pressure Safety Injection (HPSI) System circuit wiring (documented in NRC Inspection Report number 50-309/96-11), the plant was returned to cold shutdown on August 18, 1996. After the HPSI wiring was repaired and other Engineered Safeguards Features (ESF) functional tests were performed, the plant was restarted on September 1, 1996 and returned to 90% power on September 4, 1996. There was a temporary down power to 80% starting on September 10, 1996 to conduct repairs on one of the four Circulating Water (CW) System pumps. At the end of this inspection period, the plant was still at 80% power.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

Using Inspection procedure 71707, the inspectors conducted reviews of ongoing plant operations. The conduct of operations was professional and safety focused. Operators maintained the plant safely during all modes of operation.

O1.2 Reactor Shutdown and Startup

a. Inspection Scope

On various occasions, the inspectors observed operators conduct a shut down of the reactor and also a start up. The inspection activities were focused at operator performance including adherence to procedures at ensuring safe manipulation of the plant.

b. Observations and Findings

Operators conducted activities safely and in accordance with approved plant procedures. The inspectors noted that applicable procedures were used during conduct of activities and that there was proper command and control of activities. Approaches to criticality were controlled, safe and properly calculated.

c. Conclusions

Operators demonstrated good technical knowledge and safety perspective during plant operations.

O2 Operational Status of Facilities and Equipment

The inspectors observed operator actions and reviewed design documents and conducted walkdowns of portions of selected safety related systems and facilities to verify that activities were being conducted safely and that systems and facilities were being maintained well.

O2.1 Primary Component Cooling Water System

a. Inspection Scope (71707)

The inspector conducted walkdowns of portions of the Primary Component Cooling (PCC) Water System to ascertain that the system was being maintained well and capable of performing it's intended safety function.

b. Observations and Findings

The inspector inspected the pumps, various valves and sections of piping, and the surge tank. The tour included the pumps and heat exchanger areas, the emergency diesel generator PCC valves and pipes, accessible areas of the piping penetration room in the primary auxiliary building, and the PCC Heat Exchanger flow control valve area. The control room switch alignment and indications were also inspected. The valves were aligned properly and in accordance with the approved piping and instrument diagram, and the control room indications and alignment properly reflected actual system configuration. The inspector did not observe any discrepancy.

c. Conclusions

The PCC system was maintained operable and capable of performing its design function if required.

O4 Operator Knowledge and Performance

O4.1 Dropped Rods During Reactor Startup

a. Inspection Scope

The inspector observed plant startup activities in the control room. The inspector watched reactivity control and assessed operator response to startup anomalies such as dropped rods and control element assemblies (CEA) position indication problems.

b. Observations and Findings

During a reactor startup on September 1, 1996, the inspector observed control room activities and noticed some discrepancies with the CEA controls. There were instances of dropped rods and some instances of CEA position indication problems.

Rod 3, group 4 dropped from step 42 to step 3. Rod 4, group 4 dropped from step 35 to step 3 and again from step 120 to step 28. Following each rod drop, the affected group was reinserted and realigned in accordance with Abnormal Operating Procedure (AOP) 2-21, Misaligned (Dropped) CEA. Operators responded well and maintained the plant safely. This is a repetitive problem due to a design deficiency.

c. Conclusions

Operators responded well to plant anomalies such as dropped CEA rods during reactor start up.

O8 Miscellaneous Operations Issues (92700)

O8.1 Severe Weather Preparation

a. Inspection Scope

On September 1, 1996, the inspector reviewed Maine Yankee's actions taken to ensure safe plant operation when a hurricane (Hurricane Edouard) was approaching the plant area.

b. Observations and Findings

Operators reviewed Abnormal Operating Procedure (AOP) 2-40, Severe Weather, which contained the actions to be taken in preparation for a severe weather condition. The inspector reviewed the procedure and discussed the actions described in it with the operators. The procedure provided the key points to be considered such as declaration of emergency action levels, load reduction, loss of offsite power considerations, securing of structures, tanks and equipment in and around the plant.

The inspector observed that in addition to reviewing the procedure, operators were in communication with upper management to keep them abreast of the condition and to obtain guidance as required. Operators were also maintaining contact with the National Weather Bureau and Central Maine Power to track the path of the hurricane. Plant operators were being dispatched to conduct walkdowns of outside structures and equipment to ensure that things were secured.

While the AOP contained the general topics to be considered, it lacked specificity in addressing specific actions to be taken and when, to deal with actions such as specific wind speeds, staffing requirements to deal with the expected severe weather, and specific power level restrictions.

c. Conclusion

Operators took actions to ensure continued safe operation of the plant during severe weather. They reviewed the applicable procedure, and they communicated with the National Weather Bureau and Central Maine Power to properly track the hurricane. However, the procedure, AOP 2-40, Severe Weather, that contained the instructions on responding to severe weather, lacked specificity because it did not

address specific actions to be taken to deal with issues such as specific wind speeds, staffing requirements to deal with the expected severe weather, and specific power level restrictions.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

The inspector observed interaction between Operations and Instrumentation and Controls (I&C) department personnel when problems occurred during the performance of two I&C surveillance tests on September 10, 1996. During the performance of the primary calibration of the gaseous Radiation Monitoring System (RMS) Instrument, control power was lost to the RMS cabinet (Cabinet I), when the Air Ejector Radiation Monitor drawer was pulled out in the RMS cabinet. The I&C technician immediately stopped all activities and contacted his supervisor. The situation was properly discussed among the operators and the I&C personnel before any further actions were taken. Subsequent troubleshooting revealed that a rolled power cord in the cabinet had been frayed and the exposed wire had caused a short in the control power circuit, resulting in a blown fuse and loss of control power. During the troubleshooting process, operators properly verified that the automatic control functions of the monitors in the cabinet had not been lost.

The inspector also observed another situation that resulted in an unexpected control room annunciation. I&C technicians were conducting a reactor protection system (RPS) calibration (3-6.2.2.2, Reactor Coolant System flow), when a Rod Drop annunciation (R-1-6U, Power Indication, Dropped CEA) was received in the control room. The operators and the technicians immediately recognized that the alarm was unexpected relative to the RPS activities that were ongoing and stopped those activities. The situation was properly discussed and supervision from both the I&C and operations departments were involved. Maine Yankee's troubleshooting determined that the alarm was spurious in nature and not the result of an identifiable component failure. Subsquently, the RPS calibration was successfully completed.

a. Inspection Scope (62707, 61726)

The inspectors observed all or portions of the following work activities:

WO 96-2877-01, Main Steam Valve MS P-168 Repair
WO 96-02879, Residual Heat Removal Valve RH-4 Repair
WO 96-2947, Troubleshoot/Repair RPS S/G Low Pressure Trip, Channel A.
3.1.20.1. Containment Isolation Valve Testing (S/G Blowdown Valves)
3-1-2.3, ECCS Routine Testing - Valve Testing and Position Verification
3-1-22, Emergency Feedwater Flow Testing
3-1-5.3, Auxiliary Feedwater Pump, P-25B Quarterly Test

b. Observations and Findings

The inspectors noted that activities were conducted safely and in accordance with approved documents. Technicians and operators showed good knowledge of the work activities. Supervisors properly monitored work activities. Specifics of observed work activities are described below:

M1.2 Residual Heat Removal Valve RH-4

a. Scope of Inspection (62707)

The inspector observed a portion of the repair of valve RH-4 in the Reactor Containment. The repair involved a replacement of the valve after a crack in the valve body was discovered.

b. Observations and Findings

The inspector observed a portion of the repair activities for valve RH-4 located in reactor coolant loop 2 at the -2 foot elevation of the containment. Prior to the initiation of work, Contingency Plan No. 95-028 was developed and approved to address the actions necessary to be in place and to be taken during the repair to RH-4 when the residual heat removal (RHR) system would be inoperable. With RHR unavailable, the steam generators and the condenser were the only means of maintaining the RCS temperature less than 210° Fahrenheit (F) for Technical Specifications (TS) operational condition 3 (cold shutdown). If the RCS temperature exceeded 210° F, then containment integrity would be required for Technical Specifications operational condition 4 (Transthermal). Since the plant was relying on the condenser for heat removal (steam generators secondary side under condenser vacuum, steaming into the condenser and being fed by the condensate pumps) a contingency plan to address the loss of condenser vacuum was developed.

The inspector reviewed the contingency plan and discussed the plan with the licensee. The plan was detailed and reflected an excellent safety perspective, and good management involvement in activities that had the potential to significantly affect the plant.

The inspector observed the quality control (QC) inspector performing a non-destructive examination (liquid penetrant check) of the newly installed valve welds. No indications of flaws in the new welds were identified by the QC inspector.

The inspector noted that maintenance workers, supervisors, radiation control technicians and quality control inspectors properly discussed the scope and nature of the work prior to commencing work. The inspector noted that station radiation controls personnel properly controlled access to the loop 2 area and maintained proper controls to minimize the spread of contamination. The valve was freeze sealed prior to replacement to eliminate any water in the piping during the welding process.

A new valve with a section of piping was prefabricated in the shop to allow for faster installation in the field and subsequent reduction of radiation exposure to the workers. After installation, while reviewing the completed work package, the job lead maintenance personnel identified a concern with the weld wire used by the welder. The wire did not conform to the weld procedure requirements. Specifically, weld wire other than that specified by the procedure was used to prefabricate and install the new RH-4 valve, couplings, and piping.

The Maine Yankee Welding Committee was convened to review the finding and provide assessment of the concern. Upon consultation with the station Metallurgist, it was determined that the stainless steel weld wire used was appropriate for this application. The Welding Committee determined that the work package was not specific for type of weld wire required for each weld joint. The piping, coupling and valve materials were specified but the welder was left to determine the proper weld wire. The committee recommended to the Maintenance Manager that a departmental root cause determination be made on the issue.

c. Conclusions

The inspector determined that overall, the performance of maintenance was satisfactory, however it was only fortuitous that the weld wire used by the welder was appropriate. The maintenance department lead mechanic demonstrated a good questioning attitude. Maine Yankee appropriately determined that the work control process associated with the selection of weld wire was in need of improvement.

M1.3 Main Steam Valve MS P-168 Repair

a. Inspection Scope (62707)

The inspector observed a portion of work activities on Maine Steam Valve, MS P-168, and held discussions with maintenance workers and supervisors.

b. Observations and Findings

On August 13, 1996, the inspector observed Maine Yankee mechanical maintenance personnel during the repair of the main steam inlet valve (MS-P-168) for the auxiliary feedwater pump turbine. After inspection by maintenance personnel the valve was found to have steam cuts on the gasket and gasket seating surface. A work order (WO 96-2877-01) was written to repair the lower gasket seating surface and reface the seating surface to original valve vendor dimensions. The repairs consisted of grinding out the steam cut indications, rewelding and surface grinding.

The inspector observed the machining of the gasket seating surface and noted that the maintenance personnel had properly controlled access to the area and that maintenance supervisors provided good oversight of the repair activity to ensure all work control measures were being adhered to while the work was in progress.

c. Conclusions

The inspector determined that Maine Yankee maintenance personnel properly performed the repair activities as required by the station work control program. Maintenance personnel were observed to perform their tasks in a professional manner with excellent supervisory oversight in evidence at the work site.

M1.4 Reactor Protection System, Steam Generator Low Pressure Channel A Repair

a. Inspection Scope (62707)

The inspector observed a portion of troubleshooting activities of the Reactor Protection System (RPS) Steam Generator (S/G) Low Pressure Channel A, to ascertain that technicians were properly following approved work instructions and that activities did not negatively affect plant operations.

b. Observations and Findings

During a backshift inspection, the inspector observed station Instrument and Controls (I&C) technicians during troubleshooting and repair activities on the RPS S/G Low Pressure Channel A. Earlier, the channel could not be properly calibrated. Replacement of the bistable did not resolve the problem. Further investigation revealed a faulty drawer cable connector. The replacement of the connector was accomplished satisfactorily and the channel restored to operable status.

c. Conclusion

The inspector noted that the I&C technician demonstrated a persistent and thorough approach to problem resolution during the process. He displayed good analytical skills to resolve the problem in a controlled and safe manner.

M1.5 3.1.20.1. Containment Isolation Valve Testing (Steam Generator Blowdown Valves)

a. Inspection Scope (61726)

The inspector observed a portion of the performance of surveillance test 3.1.20.1, CI Valve Testing at Power. The inspector observed the stroke testing of the steam generator blowdown isolation valves, BD-T-12, 22, and 32.

b. Observation and findings

The test was required to be conducted on a quarterly basis while the plant is at power in accordance with Technical Specification 4.6, Periodic Testing, to verify that the valves can perform their safety function.

The inspector found the test activities to be well controlled and conducted by the operations crew. The testing was conducted in accordance with the approved procedure. The stop watch used was within its calibration frequency. Operators followed test instructions including the applicable "Precautions" contained in the procedure. The three valves were tested satisfactorily.

c. Conclusions

The inspector noted that test activities were conducted well by knowledgeable and capable individuals. The Steam Generator Blowdown Isolation Valves were properly tested and verified operable.

M1.6 3-1-2.3, ECCS Routine Testing - Valve Testing and Position Verification

a. Inspection Scope (61726)

The inspector observed a portion of surveillance test 3-1-2.3, ECCS Routine Testing - Valve Testing and Position Verification. The inspector observed the stroke testing of the High Energy Line Break (HELB) Isolation Valves, BD-T-141, BD-T-142, BD-T-143, BD-T-144, BD-T-145 and BD-T-146.

b. Observation and findings

The test was conducted in accordance with Technical Specification 4.6, Periodic Testing, to verify the operability of the emergency core cooling system's (ECCS) remotely operated valves. Test activities were well controlled and conducted in accordance with the approved procedure. A discrepancy was identified with valve BD-T-146 involving a faulty position indication. When the valve was stroked to the open position, the green (closed) indication light remained lit after the valve had been locally verified to be open. A work order was written to repair this deficiency. Operators were able to locally verify that the valve was performing satisfactorily. The inspector reviewed the licensee's actions and plan to address this discrepancy and was satisfied that it was being well addressed. No other discrepancy was noted.

c. Conclusion

Test activities were conducted well by knowledgeable and capable individuals. The HELB valves were properly tested and verified operable.

M1.7 3-1-22, Emergency Feedwater Flow Testing

a. Inspection Scope (61726)

The inspector observed a portion of Technical Specification required Emergency Feedwater flow surveillance test as directed by station procedure 3-1-22, Emergency Feedwater System Cold Shutdown Flow Test.

b. Observations and Findings

The test was performed to satisfy the requirements of Technical Specifications 4.6.B and 4.7, the Maine Yankee Inservice Testing (IST) program and Appendix R requirements. The purpose of the test was to verify the proper transfer of water from the Demineralized Water Storage Tank to each steam generator.

The testing was performed by a licensed reactor operator from the Alternate Shutdown Panel (ASP). A station nuclear plant operator (NPO) monitored the pump during the runs from the emergency feedwater pump room. The NPO performed the initial valve and switch lineups prior to commencement of the flow testing. During the period of observation, emergency feedwater pump P-25A was started twice from the ASP, but due to difficulties in obtaining the required flow measurements from the special test equipment (Clamp-on device) the testing was aborted until plant engineering personnel could identify and resolve the flow measurement problems. Plant engineering personnel relocated the flow measuring device to a more appropriate location and were able to obtain satisfactory flow measurement.

c. Conclusions

The inspector determined that Maine Yankee operations personnel properly adhered to the surveillance procedure and proactively terminated the test when the flow anomalies were observed. The operators kept constant communications with the control room during the testing to ensure the testing was conducted safely.

M1.8 3-1-5.3, Auxiliary Feedwater Pump, P-25B Quarterly Test

a. Inspection Scope (61726)

The inspector observed portions of the quarterly surveillance test of Auxiliary Feedwater Pump, P-25B to ascertain that it was conducted in accordance with approved procedures and that it properly demonstrated the operability of the pump.

b. Observations and Findings

On September 5, 1996, the inspector observed a portion of the quarterly surveillance testing of the auxiliary feedwater pump (P-25B). The test was performed to provide additional data concerning the pump performance after recently being placed in Maintenance Rule A1 category. The test was controlled by station procedure 3-1-5.3, "Auxiliary Feed Pump, P-25B, Test." A nuclear plant operator was stationed at the pump to monitor the operating parameters to ensure proper operation. In addition, plant engineering department personnel took additional readings on the pump for reliability data for trending purposes to monitor pump performance.

Upon starting, the pump operated smoothly with no indications of excessive vibration or flow oscillation. However, the inspector did note that the NPO had to physically hold gage TI-1111B (local bearing oil temperature indicator) to get a

stable reading. When questioned by the inspector, the NPO stated that this temperature indicatcr had been a problem for a considerable amount of time. The inspector discussed the problem with the plant Shift Supervisor, who agreed to investigate the problem. A station work order (WO 96-03436) was generated to investigate and repair the cause of the problem with the temperature indicator.

The test was completed satisfactory with no other identified concerns.

c. Conclusions

The inspector determined that Maine Yankee operations personnel with assistance from plant engineering personnel properly conducted the required surveillance testing of the auxiliary feedwater pump. With the exception of one minor problem, a vibration of the temperature gage, the test was successful in demonstrating the pump's capability to provide water to the steam generators when required. The inspector considered the vibration problem of the temperature gauge to be an example of an operator work around. In the NRC's Independent Safety Assessment Team Report, the number of operator work arounds was identified as an area for improvement.

M3 Maintenance Procedures and Documentation

M3.1 Pressure Switches PS-1750A and PS-1750B Calibration Records

a. Inspection Scope (61725)

The inspector reviewed the calibration records for the Secondary Component Cooling (SCC) Water System suction header pressure switches, PS-1750A and PS-1750B. The review was conducted to verify that the pressure switches were properly calibrated and maintained to ensure that they would perform their safety function.

b. Observations and Findings

Since the SCC system provides cooling water to both safeguards and non-safeguards equipment, under certain circumstances such as a seismic event, the non-safeguards load would need to be isolated. SCC valves SC-A-460 and SCC-A-461 function to isolate the non-safeguards load. These valves shut on a low suction header pressure as sensed by pressure switches PS-1750A and PS-1750B (Trains A and B respectively). If the suction pressure in the 16 inch suction line common to both SCC pumps drops to 5 psig (indicative of a pipe rupture), both pressure switches actuate, causing the two butterfly valves to isolate the non-safeguard loads in the SCC system.

The inspector reviewed the records for the calibration performed in September, 1995, for both switches. The as-found setpoint for PS-1750A was 95.06" H_2O and for Ps-1750B was 93.91" H_2O. The designed setpoint for both switches was 95.3" H_2O (5.0 psig) $\pm$ 6.93" H_2O. Therefore, both switches were found within the

required calibration and needed no adjustment. The calibrations were conducted in accordance with approved procedure 6-30-4.2, Instrumentation and Control Preventative Maintenance Activity Procedure, and the activity forms were properly completed and approved. Test equipment used were within their calibration frequencies and properly listed on the activity forms to ensure that the calibration of the pressure switches was traceable to the National Institute of Standards and Tests (NIST). The inspector identified no discrepancy.

c. Conclusion

The inspector verified that the SCC pressure switches, PS-1750A and PS-1750B, were properly calibrated and maintained to ensure that they would perform as designed in a design basis seismic event to actuate valves that would isolate the non-safeguards load of the SCC system. The calibration records for the instruments were maintained as required and provided traceability to the NIST.

M3.2 Emergency Diesel Generator Room Exhaust Fan Thermostat Calibration

a. Inspection Scope (61725)

The inspector reviewed the calibration of the emergency diesel generator room exhaust fan (FN-20A and FN-20-B) thermostats. The inspection was to ascertain that the fans would operate as required to maintain the room temperature within the acceptable limit for the emergency diesel generator operability.

b. Observation and Findings

During the NRC's Independent Safety Assessment Team (ISAT) inspection, the team identified that the thermostats for fans FN-20A and FN-20B were not routinely calibrated. The fans are required to be operable for the emergency diesel generators to be considered operable. The licensee indicated that while not calibrated, the thermostats were routinely verified operable during their monthly surveillance testing of the emergency diesel generators.

To correct this inadequacy, the licensee calibrated the thermostats in accordance with procedure 6-30-4.2, Instrumentation and Control Preventive Maintenance (PM) Activity Procedure. The inspector reviewed the calibration records. The thermostats were calibrated to actuate the fans on low speed at a room temperature of 75°F and on high speed at a temperature of 95°F. The as-found setpoint for both thermostats were within the acceptable tolerance for both the low and high speed settings. The test equipment used was within its calibration test interval.

This activity was performed well and in a timely manner once the discrepancy was identified. The inspector identified no other discrepancy.

c. Conclusion

Maintenance personnel accomplished the calibration activity timely and in accordance with approved procedure. Although not formally calibrated in the past, the thermostats were functionally tested on a monthly basis and the as-found settings indicated that they were within the required calibration. However, the failure to maintain proper calibration of equipment important to safety was an apparent violation of 10 CFR 50 Appendix B, Criterion XI, Test Control, which requires that a test program to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures.
(Apparent Violation of 10CFR 50, Appendix B, Criterion XI, see also Engineering Section)

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 Containment Spray Building Fans FN-44A and FN-44B Dampers

a. Inspection Scope

The inspector reviewed Maine Yankee's actions taken to address the concern with the potential for inlet vane dampers (VP-A-56 and VP-A-57) for the Containment Spray Building fans FN-44A and FN-44B to fail shut on a loss of instrument air.

b. Observations and Findings

On August 2, 1996, Maine Yankee engineers initiated a design basis screen (DBS No. 96-051) to address the discrepancy with the inlet dampers for Fans FN-44A and FN-44B. The primary concern was that upon a loss of non-nuclear safety related instrument air, the dampers (VP-A-56 and VP-A-57) may fail shut rendering the fans incapable of performing their safety function. The fans provide ventilation to the Low Pressure Safety Injection (LPSI) and Containment Spray (CS) pumps and Heat Exchangers area in the Containment Spray Building (CSB).

As described in the Final Safety Analysis Report (FSAR), section 9.13.2.3, the CSB exhaust system is designed to remove more than 10,000 cfm of air from the pumps area, with each fan, FN-44A or FN-44B, capable of removing the full 10,000 cfm of air. This air flow is necessary to provide cooling for the LPSI and CS pump motors when they are operating. Therefore, in a design basis accident when the pumps are called upon to operate, a loss of the non-safety related instrument air would cause dampers VP-A-56 and VP-A-57 to fail, and a subsequent failure of Fans FN-44A and FN-44B, and would result in an inadequate air supply for cooling the pumps' motors.

Upon being notified of this concern operators initiated an Unusual Occurrence Report (UOR No. 96-069). The fans were immediately declared inoperable, making the LPSI and CS pumps (hence systems) inoperable also. In a previous technical specification 3.6, Emergency Core Cooling and Containment Spray Systems, interpretation, operators had determined that FN-44A and FN-44B are required to be operable as part of the subsystem of the LPSI and CS pumps. This issue was not of an immediate safety impact at the time because the plant was in cold shutdown and the LPSI and CS systems were not required to be operable. The condition was reported to the NRC in accordance with the requirements of 10 CFR 50.72, as a condition that alone could have prevented the fulfillment of the safety function of systems that are needed to mitigate the consequences of an accident.

As an immediate corrective action, a temporary modification was designed and installed on the dampers. The temporary modification involved disconnecting the actuators for the dampers and installing locking bolts/nuts to block the dampers open. The inspector reviewed the temporary modifications (96-28-1, 96-28-2, 96-28-3, and 96-28-4) and verified that they had been written, approved and installed in accordance with the appropriate procedure (0-14-2, Temporary Modification Control). The 10 CFR 50.59 determination and the Unreviewed safety Question Evaluation were performed well and properly documented. A technical evaluation (TE No. 140-96) was completed to provide engineering justification that while blocking the dampers in the full open position eliminated the concern of them shutting upon a loss of instrument air, it will not adversely affect the spray building exhaust system operation and hence operability of the LPSI and CS pumps. The technical evaluation also addressed the effect of the maximum air flows on the filtration system and concluded that the filters would remain intact at such flows.

The inspector reviewed the work orders used to control the installation of the temporary modifications (WO No. 96-02784-00 and 96-02785-00). They were detailed, approved and included the proper 10 CFR 50.59 screening.

c. Conclusion

The inspector concluded that Maine Yankee had taken good actions to ensure that a failure of a non safety related system would not negatively impact a safety system. However, until August 3, 1996, the plant had been operated with this problem. The past operation with the potential for the dampers to fail into a position that would have prevented the fulfillment of the safety function of the LPSI and CS systems has been inadequate. This is an apparent violation of 10 CFR 50, Appendix B, Criterion III, Design Control, which requires in part that design control measures shall provide for verifying or checking the adequacy of designs (**Apparent Violation of 10CFR 50 Appendix B, Criterion III**).

E2.2 Component Cooling Water System Flow Control Valves Test

a. Inspection Scope (37551)

The inspector reviewed the results of the tests performed by members of the plant engineering department to verify that the Primary and Secondary Component Cooling (PCC and SCC) Water System valves, PCC-T-19 and SCC-T-24, function properly to ensure adequate flow to the PCC and SCC heat exchangers during a design basis accident.

b. Observations and findings

As a result of questions raised by the NRC's Independent Safety Assessment Team, the licensee developed special tests for the PCC and SCC flow control valves, PCC-T-19 and SCC-T-24. The valves function to ensure that enough cooling water flow is provided to the PCC and SCC heat exchangers during a design basis accident to provide adequate heat removal from safety related systems such as the emergency core cooling and diesel generator systems.

Work Order (WO) No. 96-02951-00, Rev. 000, Verification Testing of Valve PCC-T-19 Full Open Position Capability, was developed to test PCC valve PCC-T-19. The valve is the inlet valve to the PCC heat exchangers and is required to go up to full open to provide maximum cooling water flow to the heat exchangers. The test was conducted to verify that PCC-T-19 goes full open when the heat exchanger outlet flow temperature reached 118° F or upon a loss of instrument air to the valve. PCC-T-19 is part of a two-valve temperature control valve (TCV-3440) system and is connected via a valve linkage to PCC-T-20. The bypass flow around the heat exchangers is through PCC-T-20. When PCC-T-19 goes full open, PCC-T-20 goes full closed and vice versa.

WO No. 96-02952-00, Rev. 000, Verification Testing of Valve SCC-T-24 Full Open Position Capability, was developed to test SCC valve SCC-T-24. This valve is the inlet valve to the SCC heat exchangers and operates in tandem with the heat exchanger bypass valve, SCC-T-23. When SCC-T-24 goes full open, SCC-T-23 goes full closed and vice versa.

The inspector reviewed the test results and held discussions with test personnel. The test demonstrated that valves PCC-T-19 and SCC-T-24 opened as required to maintain flow to the PCC and SCC heat exchangers.

c. Conclusions

The functionality of the component cooling system valves, PCC-T-19 and SCC-T-24, was properly demonstrated. Plant engineering department personnel showed good technical capability at developing and conducting the tests. The inspector noted that this reflected a good support to operations being provided by the engineering organization. However, the previous absence of such test results that properly demonstrated the functionality of PCC-T-19 and SCC-T-24 appeared to be

an inadequacy in the test program. This previous inadequacy constitutes an apparent violation of 10 CFR 50, Appendix B, Criterion XI, Test Control which required that a test program to assure that all testing requirements demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures. **(Apparent Violation of 10CFR 50, Appendix B, Criterion XI, see also Maintenance Section)**.

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 Unplanned Extremity Exposure (URI 50-309/96-09-01)

a. Inspection Scope (83750)

A preliminary review was performed to evaluate the details associated with an unplanned extremity exposure of 6.954 rem that was reported to have occurred on April 16, 1996, during the change out of a filter on a radwaste processing system. Inspectors gathered information through discussions with licensee personnel on-site, and during an NRC Management meeting conducted in the NRC Region I office on August 9, 1996.

b. Observations and Findings

On July 11, 1996, the radiation protection staff received the results of an extremity thermoluminescent dosimetry (TLD) read which indicated that an individual received a dose of 6.954 rem to the left extremity (hand). The annual extremity dose limit is 50 rem and this dose represents approximately 14% of the annual limit. An extremity dose of this magnitude had not been planned, and as a result, an investigation was initiated. The worker and the worker's immediate supervisor were immediately notified, the worker was restricted from the restricted area (RA), radiological information report (RIR) number 96-010 was initiated, and all primary liquid filter change outs were temporarily suspended.

The licensee's investigation included a review of records including dosimetry, radiological surveys, and radiation work permits; a review of current work practices regarding the use of extremity dosimetry; interviews with cognizant personnel including the individuals who performed the filter change out; a review of Maine Yankee's filter handling experience; the performance of filter change out mockups; a review by the Nuclear Safety Assessment Review Committee (NSARC); and an independent review by industry experts.

The preliminary investigation revealed that the TLD phosphor was not defective, and the only time the left extremity badge was worn was during an April 16, 1996, radwaste filter change out. Dosimetry results for the individual were as follows.

- whole body TLD (for the entire quarter): 0.116 rem

- self reading dosimeter for the filter change out: 0.010 R
- digital alarming dosimeter for the filter change out: 0.004 R
- right finger ring TLD for the filter change out: 0.000 rem
- left finger ring worn during the filter change out: 6.954 rem

Although the exact origin of the unplanned extremity exposure had not been established, based on the initial review, the licensee preliminarily concluded that the extremity exposure most likely resulted from contact with a hot particle during the April 16, 1996, radwaste filter change out. This could have occurred during filter removal and disposal, during cleaning of the filter housing seating surface, or during filter installation.

The process for changing out the radwaste filter was relatively straight forward, and involved isolating the system, opening a filter housing, removing and bagging the spent filter, replacing the filter, closing the filter housing, and placing the system back in-service. The work was performed using a radiation work permit (RWP); a senior health physics technician provided constant coverage for the work; outer rubber gloves were changed frequently, including at the end of spent filter handling; frequent radiation surveys were performed, including surveys of the filter housing seating surface before the filter change out and of the spent filter itself. Dose rates on the cloth filter bag, obtained during the filter change out, were 200 mR/h near the top of the filter and 1 R/h at the bottom of the filter. One contamination smear on the top of the filter housing read 200 mR per hour gamma and 200 mrad per hour beta. A hot particle wipe was performed at the top of the filter housing, and no discrete highly radioactive particles were identified. However, the radiological controls staff noted that the identification of discrete highly radioactive particles would have been very difficult, because overall contamination levels were very high. The rubber gloves used on the job, and/or a suspected hot particle, could not be recovered for investigation because, at the end of the job, the gloves were placed in a receptacle, and sent for laundering in accordance with routine procedures.

At the time of this review, the licensee had not completed their review of this event. Additional reviews were planned to investigate the validity of the dosimetry results; the adequacy of the current dose assessment; other exposure scenarios; and the need for changes in current practices regarding remote handling, use of extremity dosimetry, and health physics monitoring.

The inspectors noted that radiological control planning, oversight, and control of work during the filter change out appeared to be good. However, the unplanned extremity exposure of 6.954 rem indicates that radiological surveys were weak under the circumstances to evaluate the extent of radiation levels and potential radiological hazards present. The inspectors noted a discrepancy in the radiological surveys performed during the job. A radiation survey was performed on the filter housing filter seating surface, after the filter housing was opened and before personnel were allowed to handle the filter. However, after the spent filter was removed from the filter housing, no survey was performed on the filter housing seating surface prior to allowing the worker to clean the filter housing seating surface. In addition, the inspectors noted that the individuals extremity dose was

not evaluated in a timely manner in that the TLD worn during the filter change out on April 16, 1996, was not read until July 11, 1996. This delay in TLD processing and dose assessment, could potentially result in a failure to control occupational dose to the annual limit.

c. Conclusion

At the time of this review, the licensee had not completed their investigation of this incident including their review of the validity of dosimetry results; the adequacy of the current dose assessment; other exposure scenarios; and the need for changes in current practices regarding remote handling, use of extremity dosimetry, and health physics monitoring. This unplanned exposure event is unresolved, pending the licensee's completion of their investigation, and additional NRC review of licensee survey and monitoring practices. (URI 50-309/96-09-01)

R8 Miscellaneous RP&C Issues

R8.1 NRC Management Meeting 96-75:

An NRC management meeting (No. 96-75) was held with representatives from Maine Yankee Atomic Power Company, on August 9, 1996, at 10:00 a.m., in the NRC Region I - Public Meeting Room. The purpose of the meeting was to discuss the status of the radiological controls program at Maine Yankee and indications of declining performance. Specific areas discussed included performance indicators; the unplanned extremity exposure of 6.954 rem, that apparently occurred on April 16, 1996; and plans for performing additional self-assessments to effect program improvements. An outline of the Maine Yankee Atomic Power Company presentation is attached.

V. Management Meetings

X1 Exit Meeting Summary

The inspector presented the inspection results to the licensee on September 17, 1996. Those present at the meeting are designated on the list of persons contacted, enclosed with this report. The licensee acknowledged the findings presented.

X3 Management Meeting Summary

X3.1 Management Visits

A preliminary review was performed to evaluate the details associated with an unplanned extremity exposure of 6.954 rem that was reported to have occurred on April 16, 1996, during the change out of a filter on a radwaste processing system. As a result of this event, representatives from Maine Yankee Atomic Power Company were invited to attend an NRC management meeting (No. 96-75) which was held on August 9, 1996, at 10:00 a.m., in the NRC Region I - Public Meeting Room. The purpose of the meeting was

to discuss the status of the radiological controls program at Maine Yankee and recent indications of declining performance in radiological controls. Specific areas discussed included performance indicators; the unplanned extremity exposure of 6.954 rem that apparently occurred on April 16, 1996; and plans for performing additional self assessments and program improvements.

X3.2 ISAT Exit Meeting

On August 23, 1996, the NRC Independent Safety Assessment Team held a preliminary exit meeting in the information center on site. Four issues were identified as requiring resolution sufficient to support safe restart of the unit. Those issues are : (1) Engineered Safeguards Features System logic tests (NRC Inspection Report No. 50-309/96-11), (2) adequacy of net positive suction head for the Containment Spray System, specifically during recirculation actuation (licensee letter of September 16, 1996 NRC staff letter of October 17, 1996), (3) operability of the temperature control valves for the Primary and Secondary Component Cooling heat exchangers (Section E2.2 of this report), and (4) operability of the thermostat for the emergency diesel generator room fans (Section M3.2 of this report). All issues were resolved prior to the unit restart.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

P. Radsky, Chemistry Section Head
\+ J. Weast, Licensing Engineer
\+ S. Smith, Manager, Operations
+* E. Heath, Radiation Protection Manager
+* G. Leitch, VP, Operations
+* C. Shaw, Plant Manger
+* J. Connel, Manger, Technical Support
* J. Bourassa, Senior QA Engineer
* A. Capristo, Supervisor, Radiation Protection
* E. Fox, Radiation Protection Technician
+* J. Frothingham, Manager, Quality Programs Department
* L. Germer, Auditor, Operator
* M. Whitney, Senior Engineer, Quality Programs Department
\+ S. LeClerc, Section Head, QP
\+ H. Gilpatrick, Section Head, Corporate Engineering
\+ E. Soule, Manager - PED, Manager
\+ T. Gifford, CED, Assistant Manger
\+ S. Nichols, Manager, Corporate Engineering
\+ J. McCumber, Manager, Engineering
\+ J. Grant, EHS, Section Head
\+ J. Hebert, Manager, Licensing & Engineering Support
\+ J. McCann, Section Head, Licensing
\+ G. Whittier, VP, Licensing and Engineering
\+ M. Veilleux, Manager, Maintenance

NRC

\+ R. Conte, Chief, Reactor Projects Branch No.5
* W. Pasciak, Chief, Reactor Projects Branch No.4
* R. Ragland, Radiation Specialist
* J. White, Chief, Radiation Safety Branch, Division of Reactor Safety
* J. Wiggins, Director Division of Reactor Safety
*+ J. Yerokun, Senior Resident Inspector

* Denotes those present at the August 9, 1996, management meeting.
\+ Denotes those present at the exit meeting on September 17, 1996

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 61725:	Surveillance Testing and Calibration Control Program
IP 61726:	Surveillance Observation
IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
IP 71750:	Plant Support
IP 83750:	Occupational Radiation Exposure
IP 92902:	Followup - Engineering
IP 92903:	Followup - Maintenance

ITEMS OPENED, CLOSED AND DISCUSSED

Opened

Apparent Violation		(1) Failure to maintain proper calibration of the emergency diesel generator room fan thermostats contrary to 10CFR 50 Appendix B, Criterion XII Test Control. (Section M3.2)
		(2) Absence of test that properly demonstrated the functionality of PCC-T-19 and SCC-T-24 appeared contrary to 10 CFR 50, Appendix B, Criterion XI, Test Control. (Section E2.2)
Apparent Violation		The past operation with the potential for the CS building fan dampers to fail into a position that would have prevented the fulfillment of the safety function of the LPSI and CS systems appeared contrary to 10 CFR 50, Appendix B, Criterion III, Design Control. (Section E2.1)
50-309/96-09-01	URI	Unplanned extremity exposure including potential for the failure to perform a radiation survey. Open pending the completion of licensee investigation of the incident and additional NRC review of licensee survey and monitoring practices. (Section R1.1)

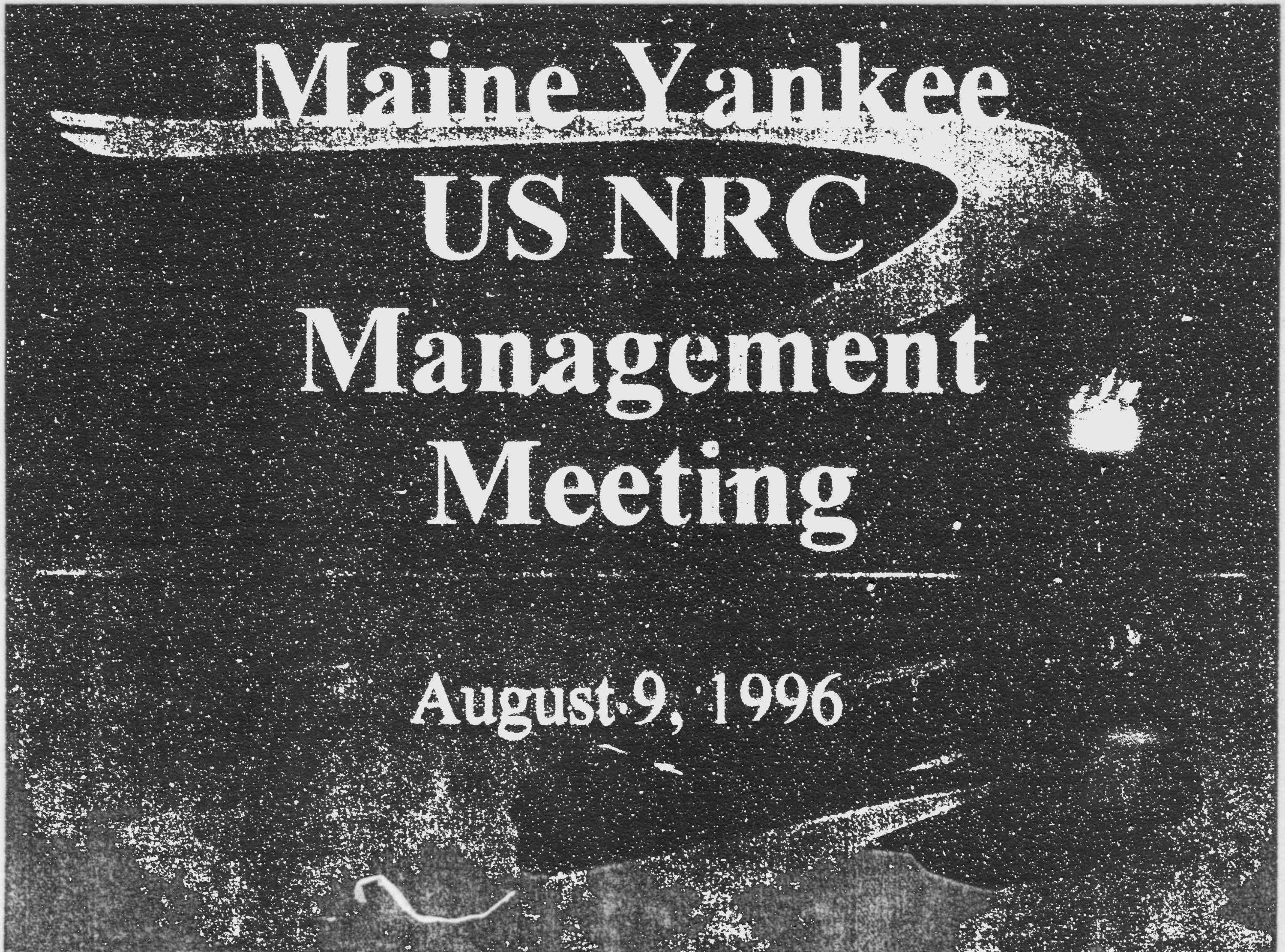
LIST OF ACRONYMS USED

AOP	Abnormal Operating Procedure
ASP	Alternate Shutdown Panel
BD	blow down
CEA	control element assembly
CED	Corporate Engineering Department
cfm	cubic feet per minute
CFR	Code of Federal Regulations
CI	containment integrity
CS	Containment Spray
CSB	Containment Spray Building
DAW	dry active waste
DBS	Design Basis Screen
DOT	Department of Transportation
ECCS	Emergency Core Cooling System
EP	Environmental Protection
F	Fahrenheit
FN	fan
FSAR	Final Safety Analysis Report
HELB	High Energy Line Break
I&C	Instrumentation and Control
ISAT	Independent Safety Assessment Team
IST	Inservice Testing
LPSI	Low Pressure Safety Injection
mR	milliRoentgen
mrad	millirad
NIST	National Institute of Standards and Tests
NSAR	Nuclear Safety Assessment Review
NSIC	Nuclear Safety Information Center
NPO	nuclear plant operator
NSAR	Nuclear Safety Assessment Review Committee
PM	Preventive Maintenance
psig	pounds per square inch - gauge
PSS	Plant Shift Supervisor
QA	Quality Assurance
QC	Quality Control
RA	restricted area
RCS	Reactor Coolant System
RIR	radiological information report
RHR	Residual Heat Removal
RMS	Radiation Monitoring System
RP&C	Radiological Protection and Chemistry
RPS	Reactor Protection System
RWP	radiation work permit
S/G	Steam Generator
TCV	temperature control valve
TE	technical evaluation

TLD	thermoluminescent dosimeter
UOR	Unusual Occurrence Report
URI	unresolved item
VIO	violation
WO	work order

ENCLOSURE 2

AUGUST 9, 1996 MEETING PRESENTATION


Maine Yankee
US NRC
Management
Meeting
August 9, 1996

AGENDA

- Introduction — C. R. Shaw
- Filter Change Extremity Dose Evaluation — A. Capristo
- Program Inititative Status and Performance Indicators — E. M. Heath
- Program Direction — J. M. Connell
- Concluding Remarks — G. M. Leitch

CHRONOLOGY OF EVENTS/INITIATIVES

■ Plant Shutdown (Refuel)	Jan 14, 1995
■ Upender Pit Hydrolaser Wand	Feb 11, 1995
■ RCP Rotating Element	Mar 24, 1995
■ Reactive Inspection	Mar 24-31, 1995
■ HP Stand Down (Radiologically Complex Work)	Mar 24 - Apr 11, 1995
■ Enforcement Conference	May 5, 1995
■ Start of S/G Sleeving Campaign	Jun 7, 1995
■ Public Meeting with NRC on Radiation Protection	Nov 21, 1995
■ Successful Completion of 100%	
▪ S/G Sleeving	Dec 11, 1995
■ Radiation Protection Inspection	Dec 11 -14, 1995
■ High Rad Key Violation	Dec 1995

CHRONOLOGY OF EVENTS/INITIATIVES (cont'd)

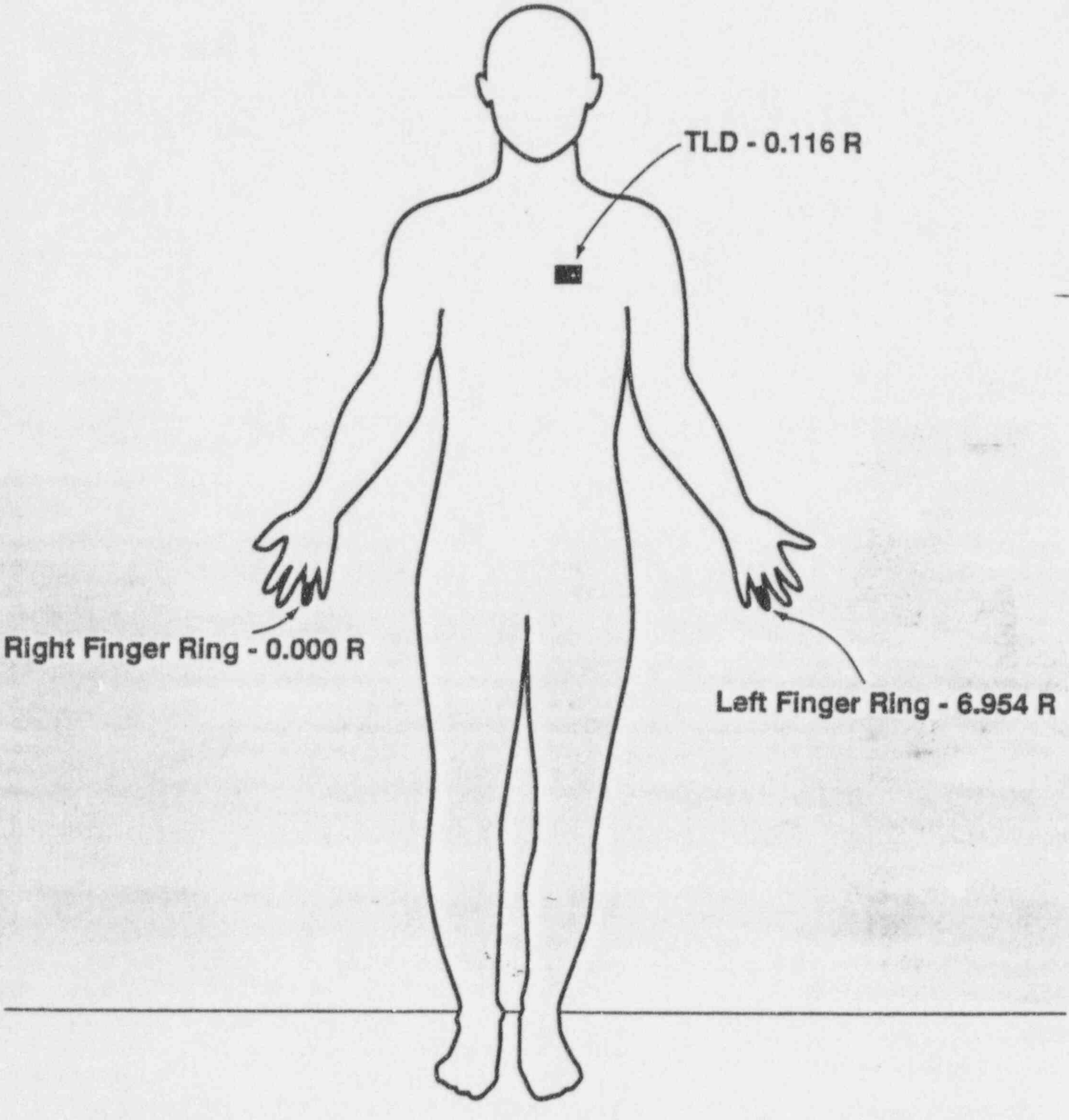
■ INPO Plant Evaluation	Mar 4-15, 1996
■ Duratek Filter Change	Apr 16, 1996
■ Mangement Meeting with NRC on CH-138, CH-99	Apr 25, 1996
■ Badge Read for 2^{nd} Quarter	Jul 11, 1996
■ Start of Spent Fuel Pool Rerack	Jun 1996
■ INPO Assist Visit	Jul 15-17, 1996
■ QPD Assessment Updates	Jun 24 ⇒ End of Year
■ NRC Management Meeting	Aug 9, 1996

FILTER CHANGE/EXTREMITY DOSE EVALUATION

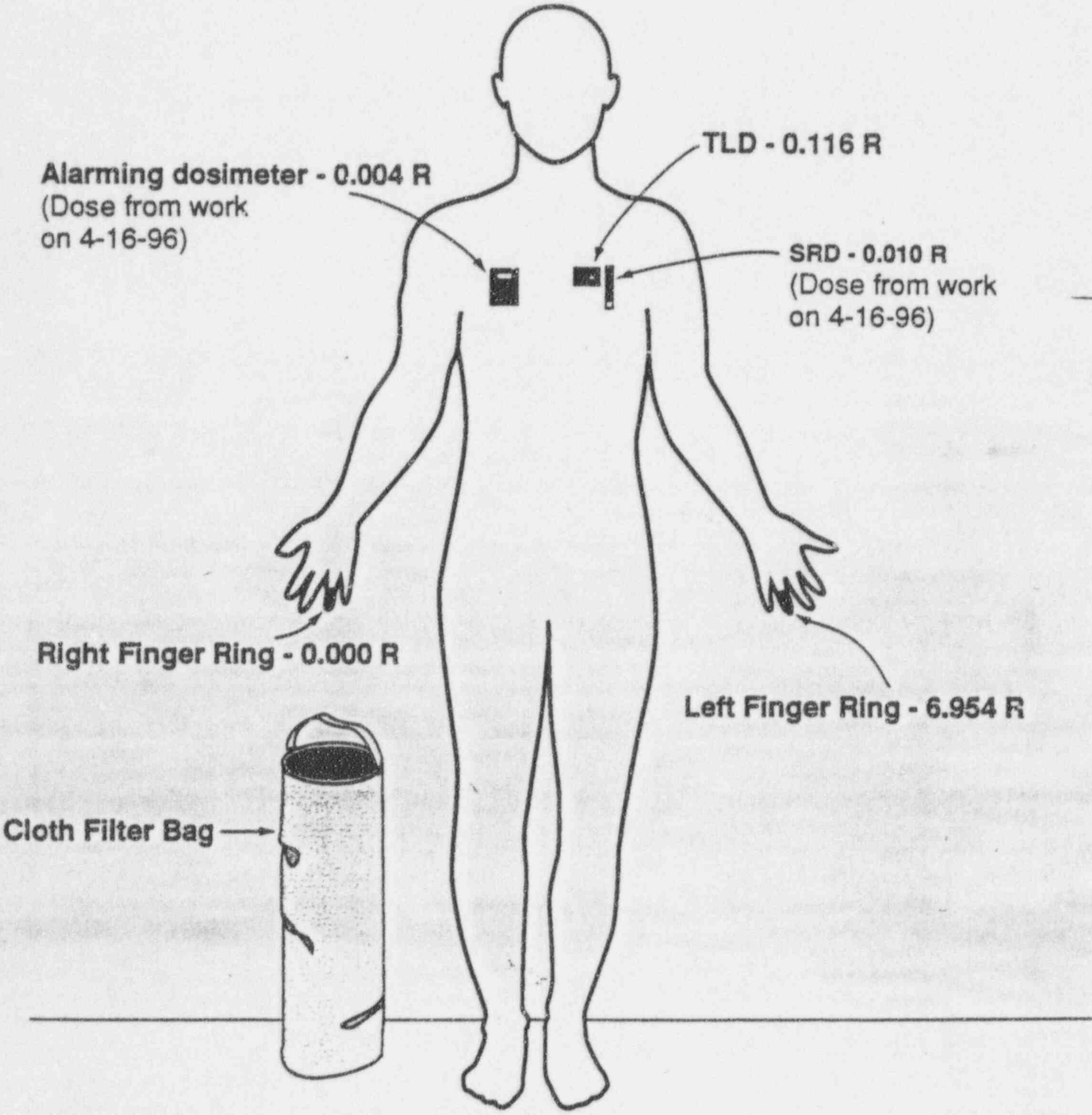
- July 11, 1996, Quarterly TLD Read Data Resulted in 6.954 rem to Workers Left Hand, 0.000 rem to the Right Hand, and 0.116 rem to Whole Body for Quarter. (Worker is Right Handed)

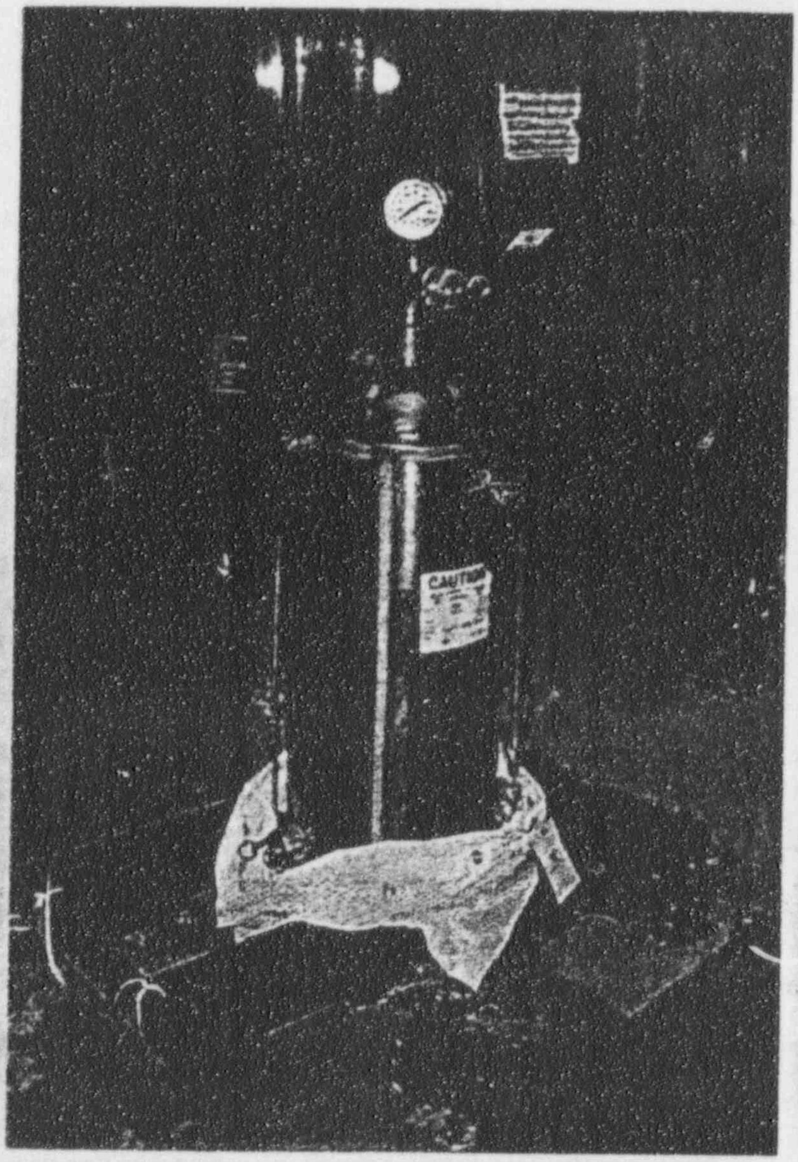
- July 12, 1996, RIR-96-010 Initiated. Worker and His Supervisor were immediately informed of concern. Worker was immediately restricted from RA. Suspended all primary liquid filter change outs.

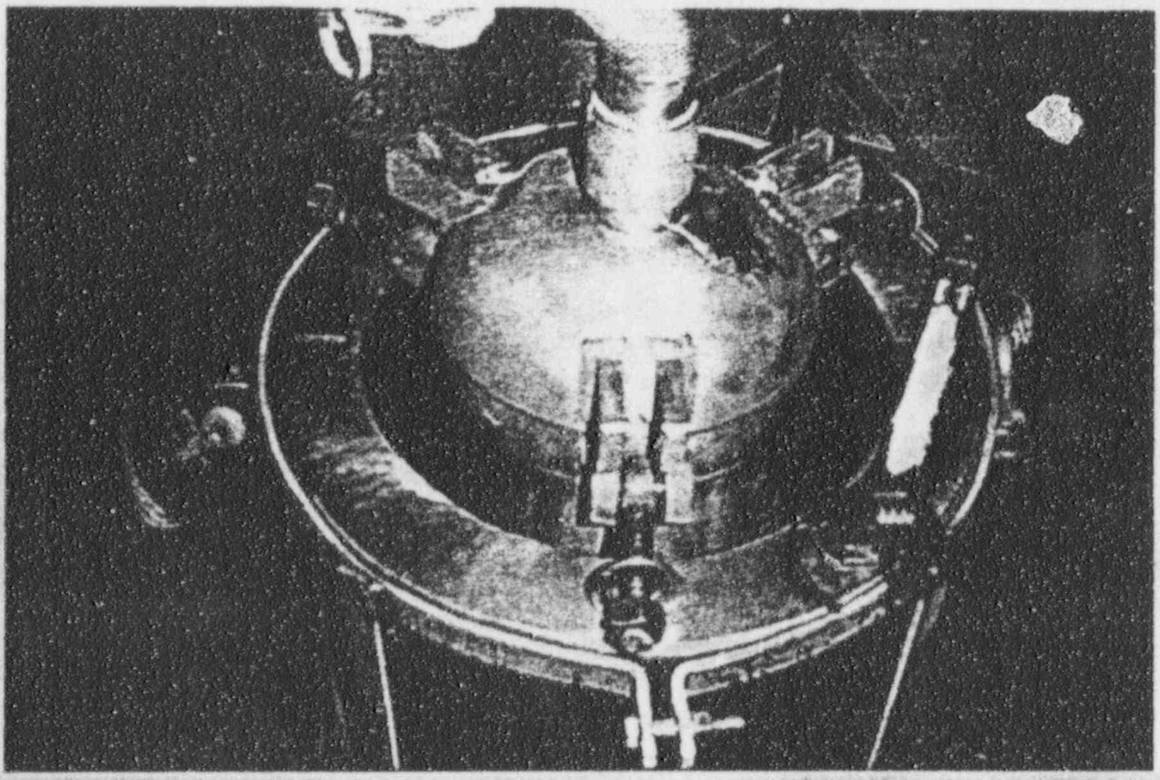
7-11-96 Data Available From 2nd Quarter Read

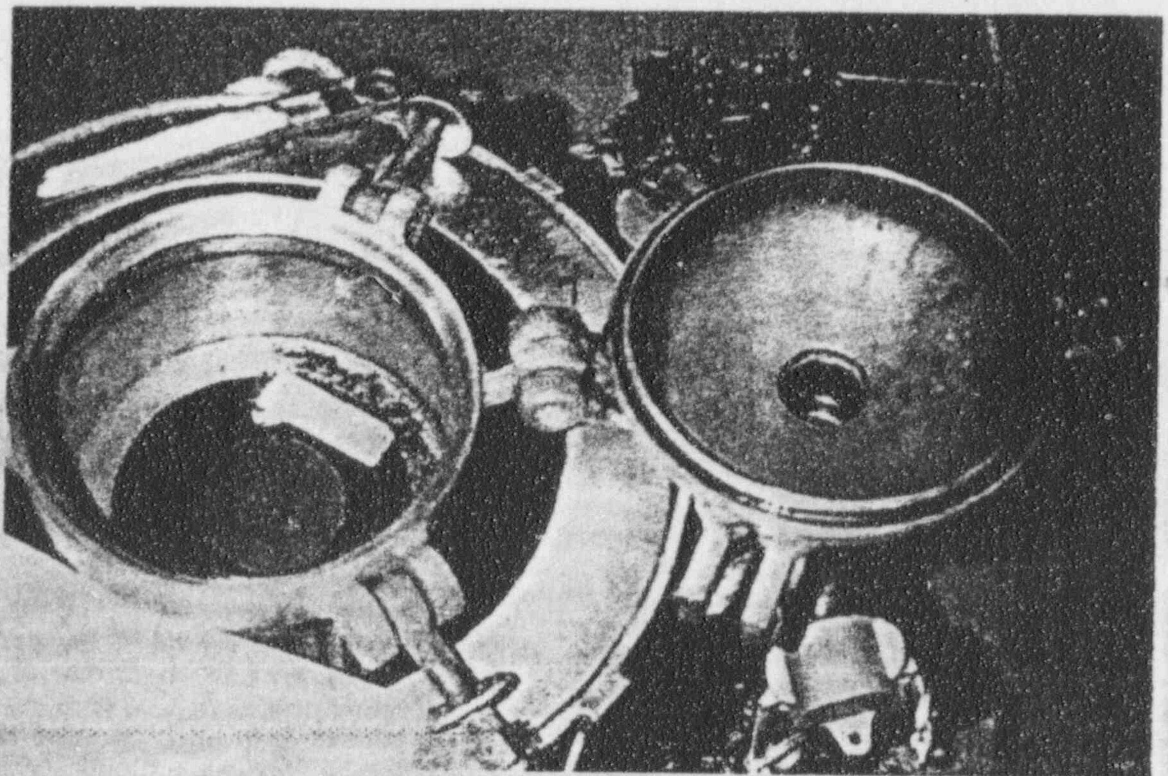


7-11-96 Data Available
After Evaluation of Work Performed in 2nd QTR

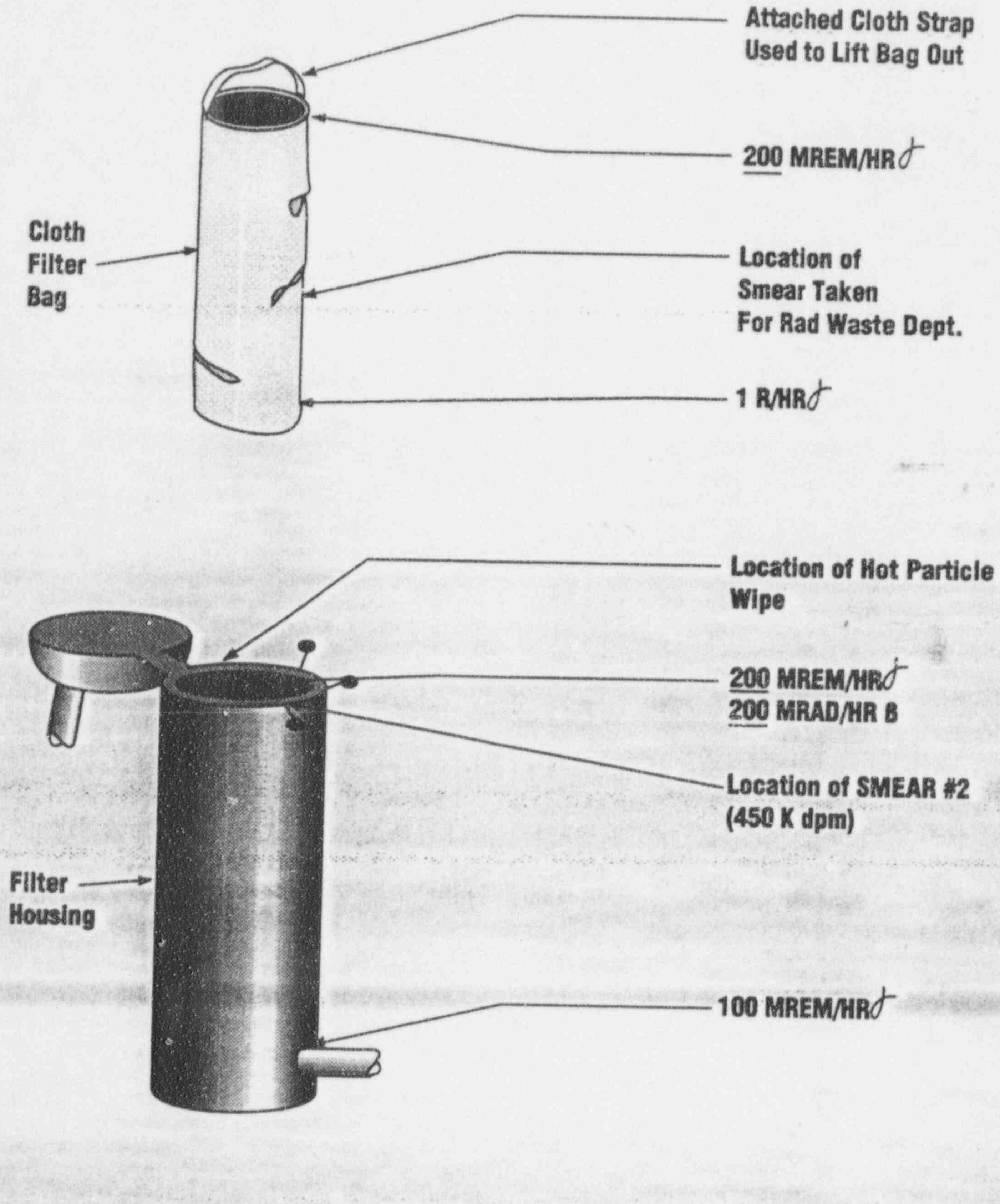








Pre-Filter


Attached Cloth Strap
Used to Lift Bag Out
200 MREM/HR γ
Cloth
Filter
Bag
Location of
Smear Taken
For Rad Waste Dept.
1 R/HR γ
Location of Hot Particle
Wipe
200 MREM/HR γ
200 MRAD/HR β
Location of SMEAR #2
(450 K dpm)
Filter
Housing
100 MREM/HR γ

EVALUATION...

- Review of extensive filter handling experience at Maine Yankee verified the non-typical nature of this level of exposure.

- April 16, 1996, change out of Duratek ion-exchange pre-filter (RWP-96-098) was worked to change "Pre" and "Post" filter. (Worked 6 times prior)

EVALUATION...

- RWP referenced the procedural guidance for evaluating the need for Extremity Dosimetry Properly. (Gradient >= 10 <u>and</u> WB dose rate >= 100 mRem/hr <u>or</u> extremity dose expected to be >= 1000 mRem with WB shielded.

EVALUATION...

- RPT opted to have the worker wear finger rings to evaluate extremity dose.
- Technician survey of open housing was completed and communicated to worker.

EVALUATION...

- Pre-filter change-out occurred in about 4 minutes.
- Remaining scope of work (replacement, closure, post-filter change, system fill, hydro, line-up) was completed in 45 minutes.

RIR FINDINGS

- RIR FINDINGS INCLUDE:
  - Procedural guidance was followed. Plant practices are comparable to industry practice (Extremity dosimetry issue, read frequency). Improvement opportunities exist.
  - Basic Contamination Control practices and Radiation Protection practices were followed.
  - Based on a comprehensive review of the exposure scenario which was reviewed by industry experts and NSAR Committee, Maine Yankee assigned worker dose of 6.954 Rem. (Investigation is not complete)

CONTINUOUS IMPROVEMENTS

- Finger rings are required on all primary system liquid filter change outs.
- Extremity dosimetry will be processed post-job, rather than quarterly.
- Evaluating current practices regarding remote handling and extremity dosimetry requirements.

FILTER CHANGE/EXTREMITY DOSE EVALUATION, continued

- Communicate heightened expectations regarding contamination/hot particle monitoring.
- Review incident with technicians
- These improvements have been captured in procedural changes and RWP modifications.
- Data obtained will be evaluated for effectiveness and permanent programmatic inclusion.

- Initiative Status
- Performance Trends
- Summary

CHALLENGES

- Continuing Events - Low Consequence
- RWP and Procedural Compliance
- High Radiation Area Controls
- Acceptance of Marginal Performance

INITIATIVES

November, 1995

- Benchmarking - Other Sites, Customer Feedback
- Continue Independent Oversight, Consultation
- RWP & Work Order Improvements
- Separation of Strategic and Tactical Centers
- Enhance Performance Indicators

BENCHMARKING

- 500 Staff Hours - Technician to Department Manager
- Initiatives:
  - Contamination Control (SFP Reracking)
  - Performance Indicators (Management Oversight)
  - Procedural Improvements (Survey Analysis)
- Senior RP Technician - INPO Plant Evaluation Team
- Customer Satisfaction Survey

Critical Self-Assessment, Thought Provoking and Accepted for Action

Implementation Meets Expectations

INDEPENDENT OVERSIGHT

- Annual External Management Assessment
- INPO Assist
- Ongoing:
  - QPD Surveillance
  - NSAR Subcommittee

Rigorous, Detailed and Challenging
Implementation Meets Expectations

RWP AND WORK ORDER IMPROVEMENTS

- Enhanced Briefings - Worker Performance
- Supervisory Support to Daily and Outage Planning
- Policy Statement
- Procedural Guidance Drafted

Needs Improved Focus -- Less Than Effective Cross-Functional Coordination

Implementation Below Expectations

SEPARATION OF TACTICAL/STRATEGIC CENTER

- RPSH Office Separated from RP Supervisors
- ALARA Office Consolidated
- Increased Supervisory Presence at Checkpoint
- Use of Strategic Status Board

Smoothed Work Flow, Enhanced Customer Interface and Issue Resolution

Implementation Meets Expectations

ENHANCE PERFORMANCE INDICATORS

- Effective Weekly Debrief at Management Meetings
- Improving Trends in Most Major Indicators
- Ongoing Evaluation to Refine

Broadened Scope and Focus Needed to Address Performance Weaknesses

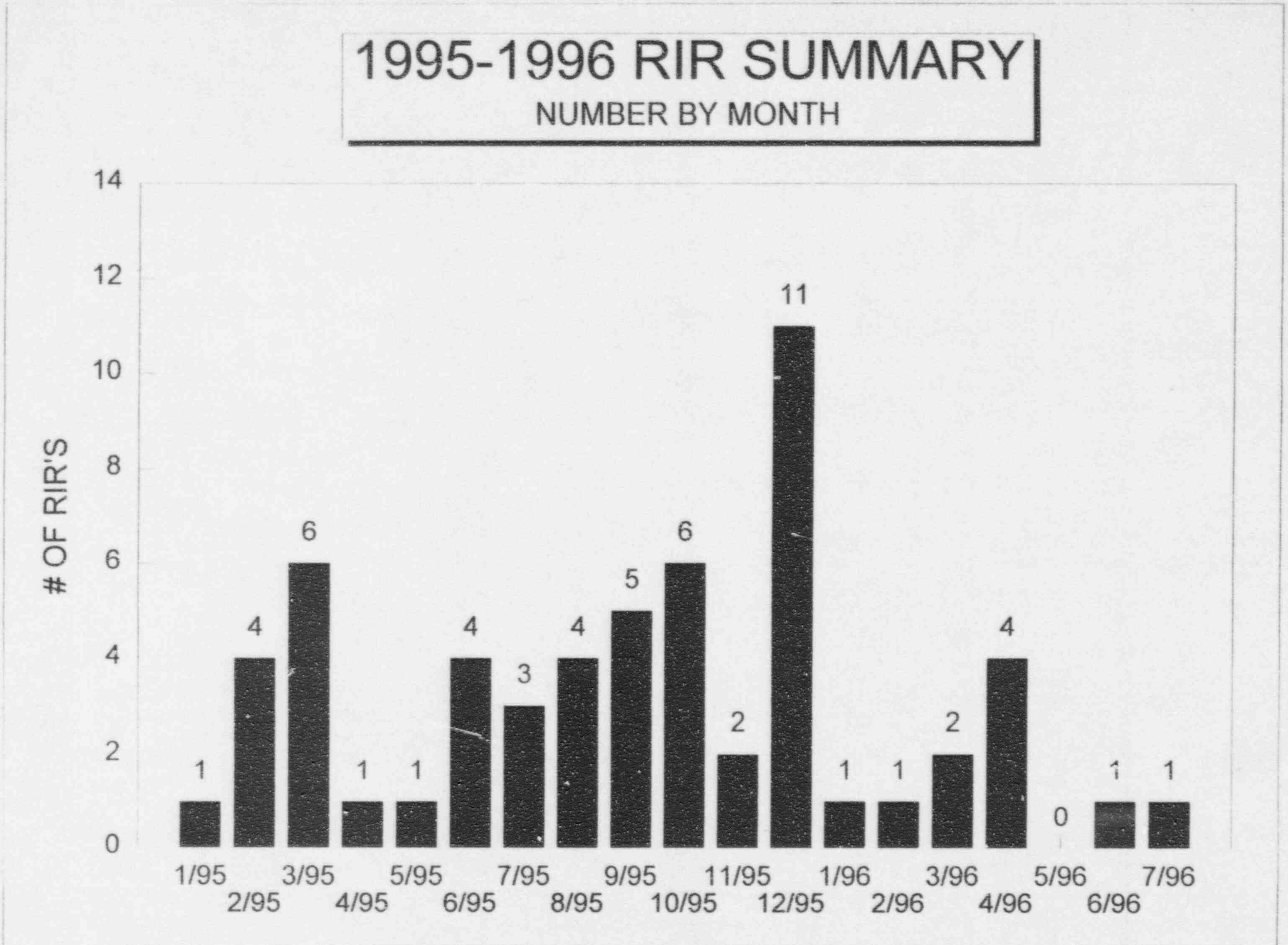
Implementation Below Expectations

PERFORMANCE TRENDS

- Radiologically Significant Work
- RIR, PWPO and PCR Data
- Trends

■ Significant Work	Person-REM	NSARC Rev
Steam Generator Resleeving	25.7	Yes
Reactor Head Installation	10.0	No
Reactor cavity upender pit decon	4.8	Yes
Refueling Mast Cylinder Replacement	1.4	Yes
Hoist Box Run-out Alignment	2.1	No
P-13a,b Containment Sump Pump Repair	0.6	No
RCP 2 Vibration Investigation	1.44	No
P-3 Repair/Upgrades	1.36	No
Containment Clean-up	6.0	No
SFP Rerack Project	2.116*	No
PDA-122/123 Repair	0.26	No
P-1-1 Oil Leak	0.45	Yes
Heat Trace	0.23*	No
Restricted Area Clean-up	3.7*	No

■ * Indicates dose for job as of 8/4/96


1995-1996 RIR SUMMARY
NUMBER BY MONTH
OF RIR'S
14
12
10
8
6
4
2
0
1
4
6
1
1
4
3
4
5
6
2
11
1
1
2
4
0
1
1
1/95
2/95
3/95
4/95
5/95
6/95
7/95
8/95
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10/95
11/95
12/95
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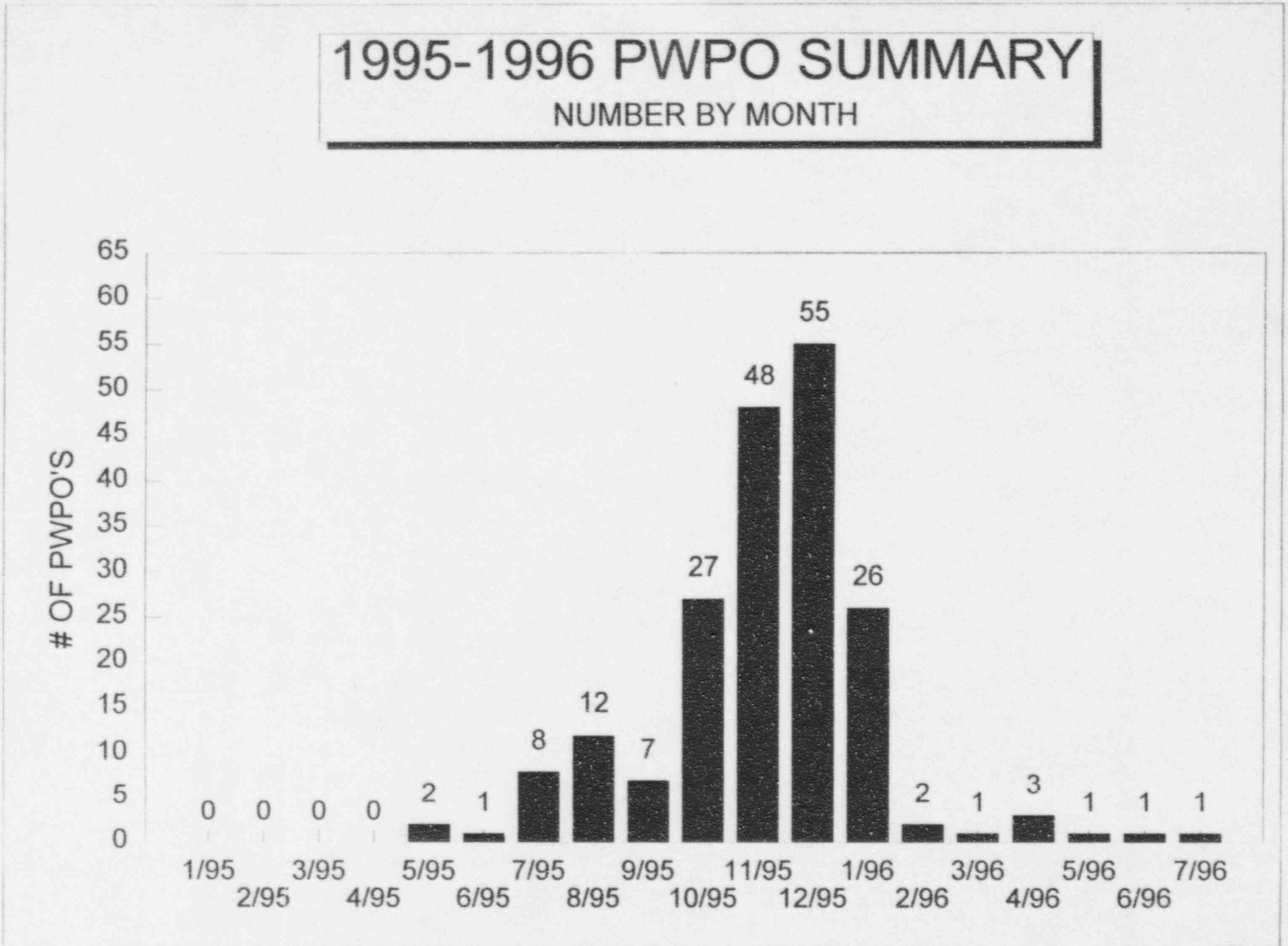
5/96
6/96
7/96


1995-1996 PWPO SUMMARY
NUMBER BY MONTH
OF PWPO'S
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60
55
50
45
40
35
30
25
20
15
10
5
0
0
0
0
0
2
1
8
12
7
27
48
55
26
2
1
3
1
1
1
1/95
2/95
3/95
4/95
5/95
6/95
7/95
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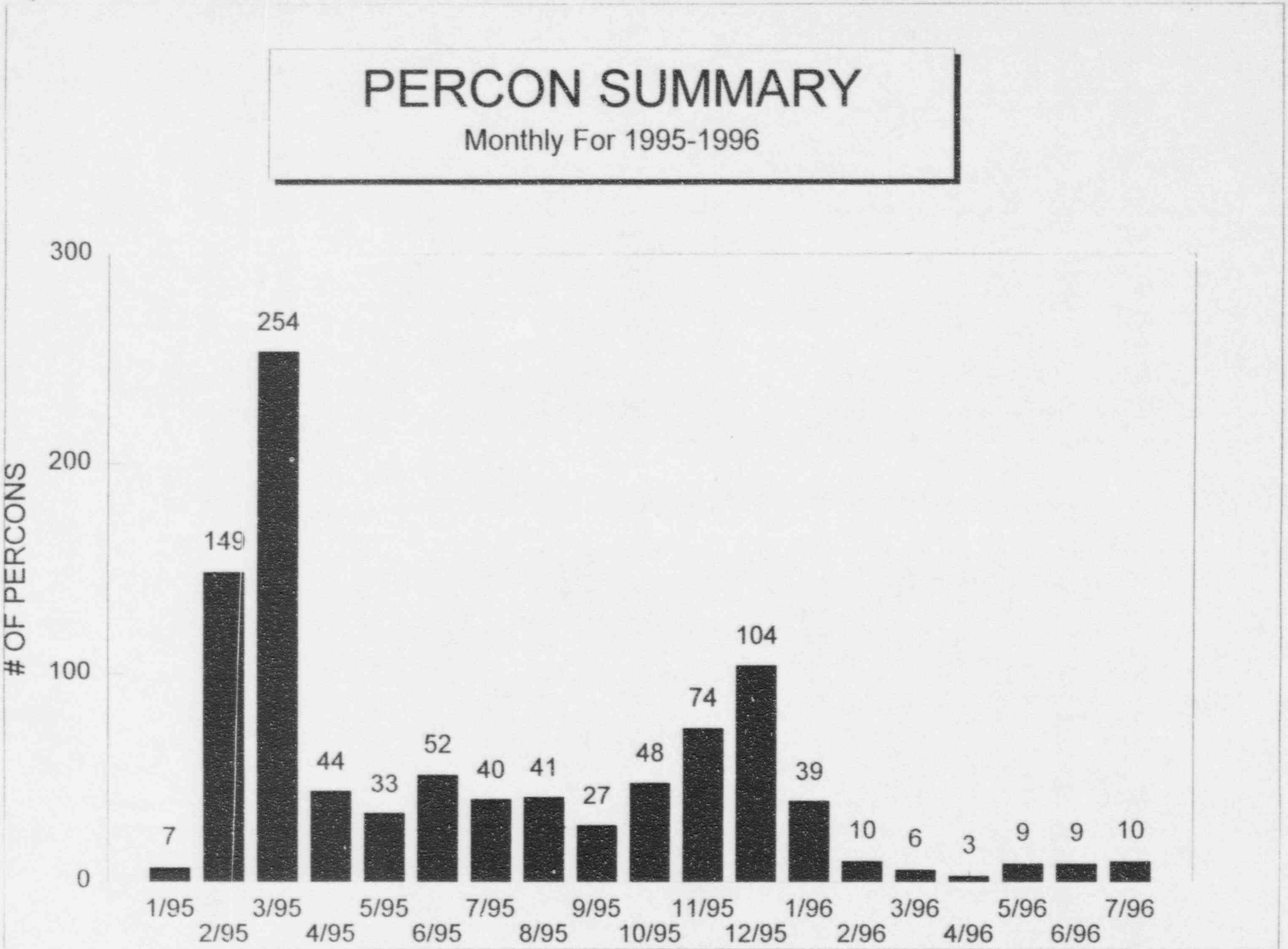
5/96
6/96
7/96


PERCON SUMMARY
Monthly For 1995-1996
OF PERCONS
300
200
100
0
7
149
254
44
33
52
40
41
27
48
74
104
39
10
6
3
9
9
10
1/95
2/95
3/95
4/95
5/95
6/95
7/95
8/95
9/95
10/95
11/95
12/95
1/96
2/96
3/96
4/96
5/96
6/96
7/96

TRENDS AND INITIATIVES

- Elimination of LHRA Door Problems
- Reduction in Posting Problems
- Line Supervision Investigation and Correction -RIRs
- Implementation of "Green is Clean"
- Barnwell Opened
- Aggressive Restricted Area Clean-Up

OVERALL ASSESSMENT

- Worker Accountability Improving
- Pace of Change Slower to Overall Improvements Than Anticipated
- Many Initiatives Begun -- Must Reign in Those Lingering to Completion
- Set Clear Priorities
- Continuing to Raise Expectations

ONGOING EFFORTS WITH MAINTENANCE ORGANIZATION

- Work Order as a Complete Tool
- Radiation Worker Responsibility
- Training
- Management of Change

OTHER SIGNIFICANT EFFORTS

- Survey Map Improvement
- Integrated Access Control/Electronic Dosimetry
- Implementation of Formal Technical Evaluation Process
- Self-Assessments
  - Program Review Process
  - Evaluation of Improved Technology
    - Health Physics Information Tracking System

CULTURAL ASSESSMENT

- Report Issued May 14, 1996
- Radiation Protection a Specific Area of Emphasis:
  - Morale and Lack of Open Communications
  - Ineffective Change Management
  - Deteriorating Organizational Relationships

CULTURAL ASSESSMENT

Continued

- **Improvement Initiatives:**
  - Skip Level Meetings
  - Daily Briefings
  - Process and Change Management
  - Supervisory Development
  - Team Building Inclusive of Entire Organization
  - Improve Cross-Functional Communication

PROGRAM DIRECTION

Program is Improving Overall

-Improvement Recognized in Independent Assessments:

--INPO

--QPD

--Also Noted in Inspection Reports

PROGRAM DIRECTION

Continued

- Rate of Improvement is not Satisfactory
- Staff Augmented with Outside Resources
  - Potential INPO Reverse Loanee
  - Assistance at Multiple Levels
- Fully Committed to Meaningful and Lasting Improvement