

UNITED STATES NUCLEAR REGULATORY COMMISSION

REGION I

Inspection Report Nos. 50-387/94-09; 50-388/94-11

License Nos. NPF-14; NPF-22

Licensee: Pennsylvania Power and Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

Facility Name: Susquehanna Steam Electric Station

Inspection At: Salem Township, Pennsylvania

Inspection Conducted: April 12, 1994 - May 23, 1994

Inspectors: G. S. Barber, Senior Resident Inspector, SSES
D. J. Mannai, Resident Inspector, SSES

Approved By:



J. White, Chief
Reactor Projects Section No. 2A,



Date

Inspection Summary: This inspection report documents routine and reactive inspections (during day and backshift hours) of station activities, including: plant operations; radiation protection; surveillance and maintenance; engineering and technical support; and safety assessment/quality verification. Findings and conclusions are summarized in the Executive Summary. Details are provided in the full inspection report.

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EXECUTIVE SUMMARY

Susquehanna Inspection Reports

50-387/94-09; 50-388/94-11

April 12, 1994 - May 23, 1994

Operations (30702, 71707, 71710)

The 'C' Source Range Monitor (SRM) was declared inoperable on May 7 as a result of a low count rate in conjunction with a low signal to noise (S/N) ratio. The calculated S/N ratio was erroneously high due to misinterpretation of a SRM count rate indication. An Event Review Team (ERT) identified that the way the information was displayed contributed to the event. The inspector noted the licensee did a credible job of analyzing the cause and proposing corrective actions prior to recommencing core reload. However, the inspector questioned why the earlier anomalous indications were not resolved prior to continuing core reload. The licensee has agreed to address this concern. Section 2.2.1 pertains.

The licensee identified a 125 VDC ground on May 9 that disabled the electrical trips for the Unit 1 'A' Reactor Feed Pump Turbine (RFPT). The inspector noted an excessive delay in manually tripping the 'A' RFPT. This item will remain unresolved pending further review. Section 2.2.1 pertains.

Maintenance/Surveillance (61726, 62703)

A five year performance discharge test for the Unit 2 'B' battery bank was completed on April 18 with indicated capacity at 83.5%. A followup test on April 26 indicated 66.6%. The vendor concluded that low float voltages resulted in marginal capacity determined in the first test. According to the vendor, the results of the second test were due to insufficient float duration following an equalizing charge. The licensee is still evaluating the vendors conclusions. Section 3.3.1 pertains.

Engineering/Technical Support (71707, 92720, 93702)

Two minor weld cracks were identified in support brackets for the #13 and #14 jet pump sensing lines. Crack propagation was limited by the installation of partial clamps that surrounded and reinforced the support brackets. The inspector concluded that licensee actions to address the sensing line support bracket cracks were conservative and based on previous industry experience. Section 4.2.1 pertains.

On May 3, the licensee discovered that Reactor Building Chilled Water (RBCW) leaked into the instrument air (IA) system due to the inadequate establishment of blocking boundaries for maintenance. The inspector noted that licensee control of recovery activities was good. A test procedure was written and implemented to flush/dry the affected air lines and approved work authorizations were written and implemented to remove/inspect/replace damaged components. Secondary containment (SC) was unaffected by the maintenance activity. The licensee used system status control to ensure SC integrity. Section 4.2.2 pertains.



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Details

1. SUMMARY OF OPERATIONS

1.1 Inspection Activities

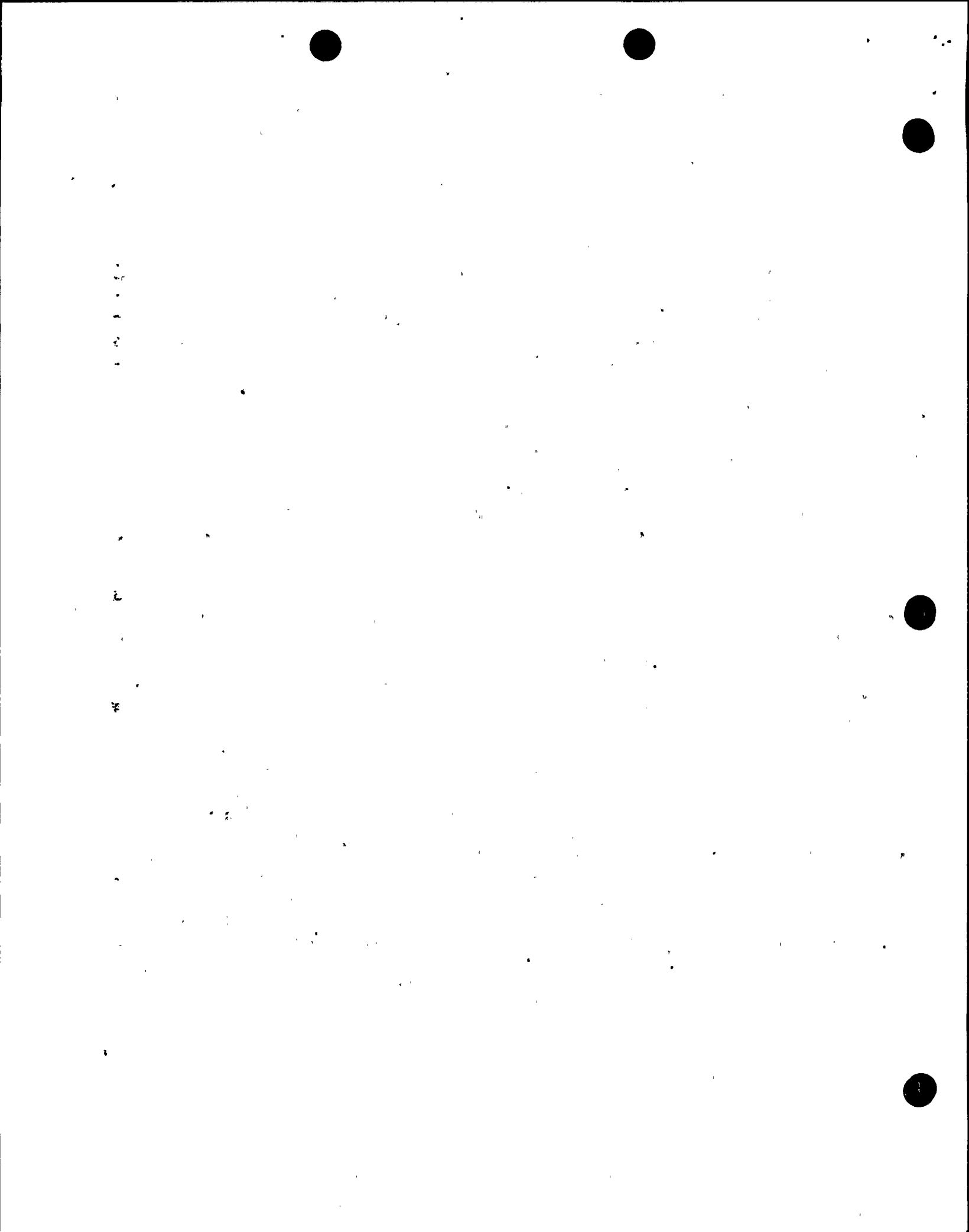
The purpose of this inspection was to assess licensee activities at Susquehanna Steam Electric Station (SSES) as they related to reactor safety and worker radiation protection. Within each inspection area, the inspectors documented the specific purpose of the area under review, the scope of inspection activities and findings, along with appropriate conclusions. This assessment is based on actual observation of licensee activities, interviews with licensee personnel, independent calculation, and selective review of applicable documents.

1.2 Susquehanna Unit 1 Summary

Unit 1 began the inspection at 100% power on May 5. On May 9, reactor power was reduced to 80% to remove the 'A' reactor feed pump turbine from service. A ground was detected in the trip logic during routine reactor feed pump turbine (RFPT) testing. The ground disabled all RFPT trips except mechanical overspeed and local manual trips. Section 2.2.2 pertains. On May 13, a non-licensed operator identified, during rounds, a pinhole leak in the emergency service water (ESW) system downstream of the 'A' emergency diesel generator (EDG) lube oil cooler. The leak was slight. The licensee took the EDG out of service and repaired the leak by installing new piping. On May 13, power was reduced to 30% for reactor recirculation motor generator (MG) set brush changeout and a control rod sequence exchange. Reactor power was returned to 100% on May 15. The unit finished the inspection period at 100% power.

1.3 Susquehanna Unit 2 Summary

Unit 2 began the inspection period in Condition 5, Refueling. Major activities included control blade changeout, control rod drive (CRD) testing, 4KV bus outages, local power range monitor (LPRM) changeout, jet pump holddown beam replacement, core reload, recirc MG set drive motor rotor replacement, and the passive water level modification installation. On April 26, the 'B' 125 VDC battery failed its discharge test. Section 3.3.1 pertains. During in-vessel in-service inspection (ISI) on May 2, a crack was identified on the #13 jet pump sensing line bracket. Additional inspection revealed that #14 jet pump sensing line bracket was also cracked. The licensee installed clamps to resolve the problem. Section 4.2.1 pertains. On May 3, Reactor Building Chilled Water (RBCW) was found leaking back into the instrument air header through check valves. Section 4.2.2 pertains. On May 7, fuel moves were halted when the 'C' SRM was declared inoperable. An operator error involving the signal to noise ratio calculation was identified. An Event Review Team (ERT) was formed. Section 2.2.1 pertains. Unit 2 finished the inspection period in Condition 5 with outage activities still in progress.



2. OPERATIONS

2.1 Inspection Activities

The inspectors verified that the facility was operated safely and in conformance with regulatory requirements. Pennsylvania Power and Light (PP&L) Company management control was evaluated by direct observation of activities, tours of the facility, interviews and discussions with personnel, independent verification of safety system status and Limiting Conditions for Operation, and review of facility records. These inspection activities were conducted in accordance with NRC inspection procedure 71707.

The inspectors performed 9.5 hours of deep backshift inspections during the period. These deep backshift inspections covered licensee activities during between 10:00 p.m. and 6:00 a.m. on weekdays, and weekends and holidays.

2.2 Inspection Findings and Review of Events

2.2.1 Unit 2 'C' Source Range Monitor Inoperable During Core Reload

On May 7, at 9:24 p.m., the 'C' Source Range Monitor (SRM) was declared inoperable in accordance with Technical Specification (TS) 3.9.2. due to a low count rate, in conjunction with a low signal to noise (S/N) ratio. These problems were related to actual hardware problems with the detector for the SRM.

The fuel and core component transfer authorization sheets (FACCTAS) requires initial verification of SRM operability after eight fuel bundles are loaded around each of four SRMs and prior to moving fuel into another core quadrant. If SRM counts are less than 3 count per second (CPS), the procedure requires operators to calculate S/N ratio to verify that it is greater than or equal to two. Earlier on day shift, after verifying all SRMs operable, operators had moved fuel into the quadrant containing the 'C' SRM. Later, on night shift, operators again verified SRM operability and concluded that the S/N ratio was out of specification for the actual conditions. Thus the 'C' SRM was declared inoperable and fuel movements were immediately suspended. Technical Specification (TS) 3.9.2 requires that at least 2 SRM channels be operable with one of the required SRM detectors located in the quadrant where core alterations are being performed and the other required SRM detector located in the adjacent quadrant. At the time, a total of 40 fuel bundles had been loaded in the core with four of them located in the quadrant with the inoperable 'C' SRM.

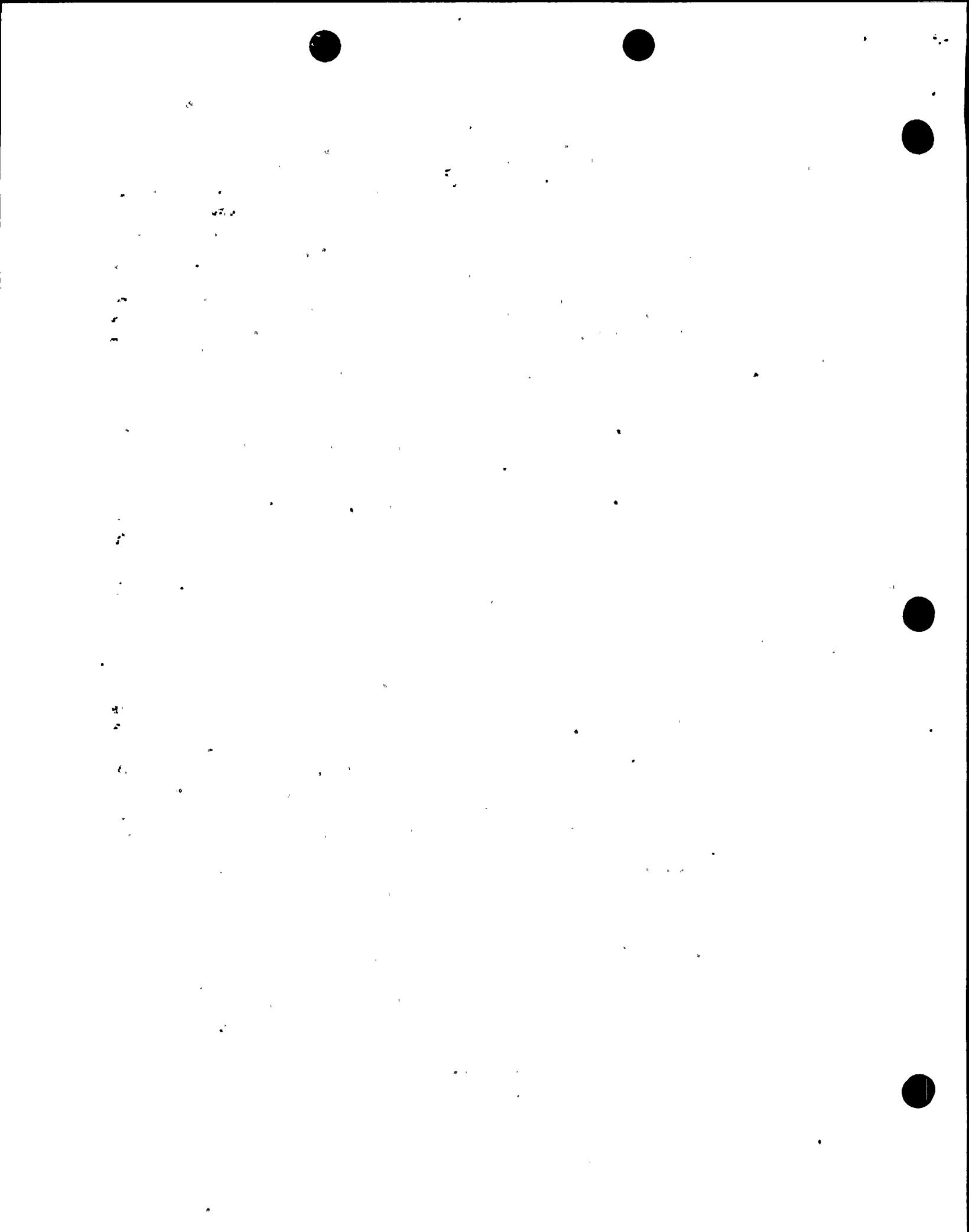
Further licensee review of the previous S/N ratio calculation identified a calculational error that masked an actual hardware problem with the SRM. The licensee noted that the displayed count rate for the actual S/N ratio was low. As a result, the licensee formed an Event Review Team (ERT) to investigate the circumstances associated with this condition.

The ERT noted that the calculational error stemmed from a poor human factors display of the scientific notation used for SRM count rate. A value of $10E-1$ (decimal value 1.0) was incorrectly interpreted as 10^{-1} (decimal value .1). This resulted in the wrong number being used for the signal to noise ratio calculation for the 'C' SRM. Consequently, the S/N ratio was incorrectly determined to be within the proper range, when it was actually beyond the acceptable tolerance for the conditions. The ERT also determined root causes and identified corrective actions. The ERT findings were presented to Plant Operations Review Committee (PORC) prior to resuming fuel load and the PORC specifically identified corrective actions which were required before resuming fuel movement. The licensee ERT identified the following causes:

- Loose 'C' SRM connector/faulty detector
- No acceptance criteria was given in the I&C surveillance procedure to define allowable differences between SRM channels
- The Display Control System (DCS) Cathode Ray Tube (CRT) SRM count rate was presented in a format different from that of the standby information panel ($10E-1$ vs. 10^{-1})
- FACCTAS directs performance of SO-200-006, Attachment I, SRM operability determination. However, it does not specify what SRM display to use for the S/N ratio calculation
- Miscommunication of the value of the count rate

The short term corrective actions (prior to resuming fuel load) included: 1) Replacement of the faulty SRM detector. 2) Operator hot box training to emphasize that use of scientific notation is not standard throughout station. 3) Procedure changes to use hardwire indication from standby information panel for SRM counts. 4) Procedure RE-081-042, FACCTAS preparation guidelines changed to direct which instrumentation to use for SRM counts. Licensee long-term actions included: 1) evaluate human factors design of CRT DCS format, 2) provide specific criteria in SRM surveillances for allowable deviation between channels when defueled, 3) determine required actions to assure future nuclear instrumentation connections are secure and consistent with equipment operating procedures, and 4) determine if operators acted in accordance with management expectations.

The inspector reviewed the ERT findings and concluded that, overall, they were thorough and comprehensive. The inspector found that appropriate short term corrective actions were implemented prior to resuming the reload. The inspector considered the PORC review of the event prior to resuming core reload a strength. PORC identified additional SOOR corrective actions and changed one corrective action from an enhancement to required action. This review ensured that appropriate actions were implemented prior to the resumption of core reload.



From review of ERT timeline for sequence of events and interviews with personnel involved, the inspector concluded that operators and technicians had questioned 'C' SRM anomalies but still proceeded with core reload until the SRM was later declared inoperable. The inspector noted that, prior to commencing core reload, the operators observed the 'C' SRM was reading high relative to the other SRMs. The 'C' SRM read 3 cps with no fuel in vessel and the A, B, & D SRM's read downscale. Also, operators noted that the 'C' SRM's period was oscillating between +100 and -100 seconds. Operators questioned these indications. Instrumentation and Control (I&C) personnel signed off the general operating procedure verifying all surveillances were complete for entry into condition 5, refueling. However, they also noted the reading was high. During initial core reload, operators noted that the 'C' SRM did not respond the same as other SRMs. The inspector noted that these operator observations were not fully resolved, yet core reload commenced and continued even though 'C' SRM response was not as expected. More rigorous follow up to these questions may have identified the 'C' SRM problems earlier. Timely resolution of these questionable indications was not addressed by either the ERT or the PORC.

The inspector considered the actual safety significance and concluded it was minimal since shutdown margin remained acceptable. However, the inspector was concerned that these anomalous indications existed and were not fully resolved prior to continuing core reload. The licensee is expected to address these matters in their report of this occurrence. The inspector will assess the licensee's actions during review of the Licensee Event Report (LER). (URI 94-11-01)

2.2.2 Reactor Feed Pump Turbine Trip Logic Ground

On May 9, at 3:45 a.m., operators were performing routine testing (OP-145-001 Section 3.12) on the Unit 1 'A' reactor feed pump turbine (RFPT). During the test, a 125 VDC ground was detected. During investigation, at approximately 5:45 a.m., the ground was found to have disabled all RFPT trips except the mechanical overspeed and local manual trip. The licensee's investigation also identified burned wires for an Amphenol connector in an inactive thrust bearing wear trip test circuit as the most likely cause of the ground. This was subsequently mentioned during shift turnover. Operators continued to evaluate the effects of the 125 VDC ground on plant operations.

Later that morning, during followup, the inspector discovered that operators consulted Technical Specification (TS) 3.3.9, feedwater/main turbine trip system actuation instrumentation to determine if the high level trip was affected by the fault. They concluded that it was not since the high level (level 8) trip would act to initiate equipment protection for all operating turbines if a high level condition existed. Although operations did not enter the LCO, they did note the difficulty in interpreting it and requested an approved written technical specification interpretation (TSI) from the nuclear compliance group. At approximately 12:00 p.m., nuclear systems engineering (NSE), I&C and electrical maintenance, while attempting to resolve the problem, recommended to operations that the 'A' reactor feed pump be secured since all automatic electric trips were disabled. Operators



These observations and/or reviews included:

- WA 30027, Local Power Range Monitor Removal, dated April 15, 1994.
- WA 46171, Instrumentation and Control (I&C) Support of LPRM Changeout, dated April 16.
- WA 30391, Control Blade Changeout, dated April 18.
- WA 34120, Install New Condensing Chamber XY-B21-2D004A, Modify Piping to Include a Vent From Condensing Chamber to Level Instrumentation Variable Leg, dated April 20.
- WA 43718, Support TP-250-004 Reactor Core Isolation Cooling (RCIC) Overspeed Test, dated May 10.

3.3 Inspection Findings

The inspector reviewed the listed maintenance activities. The review noted that work was properly released before its commencement; that systems and components were properly tested before being returned to service and that maintenance activities were conducted properly by qualified personnel. Where questionable issues arose, the inspector verified that the licensee took the appropriate action before system/component operability was declared. The following maintenance/surveillance activity required followup.

3.3.1 'B' Battery Five Year Performance Discharge Test

On April 18, the licensee performed a five year performance discharge test on the 'B' 125 VDC Battery Bank (2D620) per Technical Specification (TS) surveillance requirement 4.8.2.1.e. The purpose of the test was to verify that the battery capacity was at least 80% of manufacturer's rated capacity. After initial testing, battery capacity was verified to be 83.5%, an acceptable value. However, TS 4.8.2.1.f. requires that annual performance discharge tests be performed if capacity is less than 90% of rated capacity. Thus, to avoid the need for mid-cycle outages, the licensee chose to replace one bad cell (#39) and to clean and check the tightness of some of the other connections. The battery was recharged (equalized for 72 hours), then discharged per TS 4.8.2.1.e. on April 26, and capacity declined to 66.6%. As a result, the battery was declared inoperable per TS 3.8.2. The licensee decided to replace the battery bank with new cells.

The 'B' battery bank consists of 60 lead-calcium KCR-21 cells that were manufactured by C&D Power Systems. The KCR-21 (825 AH for 8 hours) cells were installed in 1989 to provide added capacity over the previous KCR-19 cells (742 AH for 8 hours). The licensee did not expect any battery capacity problems with this five year old battery. Typically, the capacity is guaranteed for 20 years with some small degradation expected after the fifteenth



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year. To address this concern, the licensee launched a detailed root cause evaluation, in parallel with efforts to replace the battery bank. After contacting the vendor, the licensee discovered that a replacement bank (55 cells) was scheduled to be shipped to a non-nuclear user in Georgia. The battery was diverted to Susquehanna for use, and the licensee verified the acceptability of substituting the diverted KCR-19 cells in place of the KCR-21 cells in Safety Evaluation 94-3020. Fifty-five new KCR-19 cells were added to the five remaining KCR-21 cells for the 'B' battery bank on May 5. The cells were charged between May 6 and May 8 and the performance discharge test was completed on May 12 with a battery capacity of 109%. The battery was subsequently recharged and returned to an operable status.

During their investigation and review, the licensee found that all other 125 VDC batteries (A, C, D) passed their performance discharge test with capacities of 98%, 100%, and 96%, respectively. The initial installation test for the 'B' battery bank in 1989 yielded 102% capacity. All other batteries tested in 1989 showed similar capacities. Licensee review of previous quarterly surveillances for specific gravity and voltages did not indicate a pending problem. Previous surveillance discharge tests for the 'B' battery, which require it to meet or exceed the 4 hour loss of coolant accident (LOCA) profile, were passed successfully. The inspector also questioned the licensee on whether they followed the vendor recommended voltages for float and charge operations. The licensee indicated that float and charge voltages were maintained. However, the corrected float voltages were maintained at the low end of the range.

The licensee also returned a number of cells from the first and second 'B' battery performance discharge test to the vendor for specialized testing. The vendor performed a deep discharge test on one cell of the battery from the first performance discharge test (83.5%). The deep discharge test consisted of an equalizing charge, followed by a discharge over 20 - 24 hours, followed by another equalizing charge, then a discharge for just 4 hours. The capacity of this cell was measured during 4 hour discharge at approximately 92%. As a result of this testing, the vendor attributed the licensee's poor results from the first performance test to sulphate buildup on the positive and negative plates of the battery bank. The vendor also stated that sulphate buildup on the plates could be attributable to a low float voltage. A low float voltage maintained during normal operation will allow the sulfates on plates to buildup to such a degree that the actual capacity of battery will be hindered because the electrolyte does not have ready access to the positive and negative plates.

The vendor attributed the first performance test results to the low float voltage which allowed the buildup of sulfates on the battery plates. Upon review, the licensee noted that the average individual cell voltage was typically maintained at 2.20 volts, which is at the low end of the vendor recommended range of 2.20 to 2.25 volts. The vendor attributed the second performance test results to the short duration of the float charge following an equalizing charge. The vendor noted that if float charge is too short a duration, the battery will continue to generate gas along the surface of the plates while it is being discharged; and that such gassing interferes with the discharge capacity of the battery.



The inspector observed activities, interviewed personnel, and reviewed vendor manuals and industry history to assess the adequacy of licensee actions. The inspector also reviewed the licensee's safety evaluation for the replacement of the 'B' battery bank and other calculations that supported its installation. The inspector independently calculated the discharge currents for both the first and second discharge test and found them appropriate. The licensee's initiative to contact the vendor and provide support to the evaluation of this matter was viewed as a strength. The inspector also observed the actual performance discharge test of the new battery on May 12 and noted that both the procedure and the work authorization was present and being followed by the licensee. The test rig was adequately set up and discharge current was being controlled per the procedure. No inadequacies were identified with the licensee's practices used to maintain the battery in a fully charged state. The inspector independently reviewed the individual cell voltages and noted that they trended about the battery average. Supervision was present and interacted effectively. Management involvement with problem resolution was evident. The licensee continues to evaluate the vendor's conclusions and has committed to formally track their final implementation of these or other recommendations in their internal tracking system. The inspector will review the licensee's corrective action after its completion. The inspector had no further questions.

4. ENGINEERING/TECHNICAL SUPPORT

4.1 Inspection Activity

The inspector periodically reviewed engineering and technical support activities during this inspection period. The on-site Nuclear Systems Engineering (NSE) organization, along with Nuclear Technology (NT) in Allentown, provided engineering resolution for problems during the inspection period. NSE and NT generally addressed the short term resolution of engineering problems; and interfaced with the Nuclear Modifications to schedule modifications and design changes, as appropriate, to provide long term corrective action. The inspector verified that problem resolutions were thorough and directed at preventing recurrence. In addition, the inspector reviewed short term actions to ensure that they provided reasonable assurance that safe operation could be maintained.

4.2 Inspection Findings

4.2.1 Jet Pump Sensing Line Support Bracket Cracks

On May 2, the licensee identified a crack in the #13 jet pump sensing line bracket. There are two brackets that support the sensing line, an upper and lower bracket. The weld that attached the lower bracket to #13 jet pump had a surface crack through approximately one-fourth to one-third its length. The licensee documented this nonconforming condition in NCR 94-156. The licensee inspected the sister jet pump (#3) to ascertain if additional indications existed. Since none were identified, no additional inspections were initially planned.



The licensee evaluated three options to address the weld crack, 1) a full clamp which required diffuser removal to position the clamp over the sensing line, 2) partial clamp installation which required no additional vessel disassembly, and 3) a detailed engineering analysis to justify operation for an additional fuel cycle. The licensee contacted the vendor, General Electric (GE), on May 5 to discuss these options. GE stated that the partial clamp would take approximately 1½ weeks to fabricate and can be made in parallel with doing the engineering analysis. On May 7, the licensee began core reload. On May 10, GE informed the licensee that their analysis was complete and that the evaluation required restricting low speed recirculation pump operation to no more than one hour for the entire fuel cycle.

Based on this very restrictive limit, the licensee decided to proceed with the partial cleanup replacement. On May 14, while core reload was temporarily suspended, the licensee inspected all other jet pump sensing line brackets and identified a crack in the lower bracket for #14 jet pump. Thus, another clamp was added to the fabrication order. On May 20, Safety Evaluation 94-3025, addressing the sensing line clamps for Unit 2 jet pumps #13 and #14; and ME-2RF-005, Jet Pump Sensing Line Clamps Installation Procedure, were approved by the Plant Operations Review Committee (PORC). Both clamps were installed over the next three days.

The inspector evaluated licensee actions throughout this effort including verification of bracket inspection results. The inspector also evaluated the PORC review and noted that it was thorough and complete.

The licensee plans to remove the temporary jet pump clamps and repair the affected brackets during the next refueling outage. The inspector concluded that licensee actions to address the sensing line support bracket cracks were conservative and safe based on past similar industry applications. The inspector had no further questions.

4.2.2 Water Intrusion into the Instrument Air System

On May 3, an instrument air (IA) header isolation valve (2252186) was being removed for maintenance. As the pipe was cut for valve removal, a small amount of water leaked through the pipe opening. The water came from the Reactor Building Chill Water (RBCW) system. Normally, the IA header is pressurized to 90 psig nominal with RBCW being pressurized to 30 psig nominal. The IA system also functions to automatically blow down the RB Ventilation Zone III supply coiling coils whenever freezing conditions occur. Thus, under normal conditions, any leakage at the system interface would be from IA to RBCW. In this case, with IA depressurized, RBCW leaked back through either one of two interfacing check valves (287305 or 287306). The licensee terminated this back leakage by isolating the two affected Zone III coiling coils and subsequently added these valves to the system permit. The RB sump pumps were also secured to ensure that nitrated water was not sent to the radwaste system. The inspector noted the omission of the system boundary valves from the

system permit as a major contributor to this event. The inspector also noted that the licensee agreed to review the need for proper boundary valves with personnel involved with this event.

The licensee developed a recovery plan which identified twelve IA branch lines that would have to be blown down. Additionally, specific components attached to these lines were identified for disassembly, inspection, and reassembly. To ensure proper coordination of this evolution, TP-218-014, "Flush/Blowdown I-A Header on EL 779'", was written. The TP's purpose was to perform a demineralized water flush, air blow, and moisture removal purge of entire segment of IA system effected by back leakage. The purge, blow and moisture removal purge included all of the affected piping up to the last blocking point, an IA main header isolation valve (225354).

The inspector reviewed the licensee's actions in response to this event and found that the initial actions were appropriate. The licensee's initiative in writing a TP for this evolution demonstrated an appropriate regard to control the events' complexity. The TP was very detailed and gave very specific direction for each procedure step. Some of the dampers that were wetted act to isolate secondary containment when Zone II (Unit 2 Rx. Bldg.) is connected to Zone III (Common Refuel Floor) ventilation systems. Although the TP and the work documents (WAs V43707 and V43708) did not provide any restrictions on the operation of these six secondary containment isolation dampers (HD-27524 A&B, 27576 A&B, 27586 A&B), the inspector noted that secondary containment (SC) integrity was preserved per the general operating procedure (GO-200-006) and by the system status file prior to beginning fuel movement on May 7. Thus, SC integrity was adequately maintained. The inspector had no further questions.

5. PLANT SUPPORT

5.1 Radiological Controls

PP&L's compliance with the radiological protection program was verified on a periodic basis. These inspection activities were conducted in accordance with NRC inspection procedure 71707. Observations of radiological controls during maintenance activities and plant tours indicated that workers generally obeyed postings and Radiation Work Permit requirements. No significant observations were made.

5.2 Emergency Preparedness

The inspector reviewed licensee event notifications and reporting requirements for events that could have required entry into the emergency plan. No events were identified that required emergency plan entry.

5.3 Security

PP&L's implementation of the physical security program was verified on a periodic basis, including the adequacy of staffing, entry control, alarm stations, and physical boundaries. These inspection activities were conducted in accordance with NRC inspection procedure 71707. The inspector reviewed access and egress controls throughout the period. No significant observations were made.

6. MANAGEMENT AND EXIT MEETINGS

6.1 Resident Exit and Periodic Meetings

The inspector discussed the findings of this inspection with station management throughout and at the conclusion of the inspection period. Based on NRC Region I review of this report and discussions held with licensee representatives, it was determined that this report does not contain information subject to 10 CFR 2.790 restrictions.

6.2 SALP Management Meeting

On May 17, 1994, a management meeting was held between the NRC and PP&L to discuss the most recent Systematic Assessment of Licensee Performance (SALP) report. The meeting was held onsite at the Susquehanna Access Processing Facility (APF). The meeting was open for public observation. Attachment 1 lists the meeting attendees. Attachment 2 is a copy of the NRC presentation. Attachment 3 is a copy of PP&L's presentation.

6.3 Inspections Conducted By Region Based Inspectors

<u>Date</u>	<u>Subject</u>	<u>Inspection Report No.</u>	<u>Reporting Inspector</u>
05/09/94 - 05/13/94	Power Uprate Modifications	94-10	Drysdale Whitacre

ATTACHMENT 1

SALP MANAGEMENT MEETING ATTENDEES

NAME

TITLE

Nuclear Regulatory Commission

W. F. Kane	Deputy Regional Administrator
J. Durr	Acting Deputy Director, DRP
G.S. Barber	Senior Resident Inspector, Susquehanna
D.J. Mannai	Resident Inspector, Susquehanna
C. Poslusny	Acting License Project Manager, NRR

Pennsylvania Power and Light

R. G. Byram	Senior Vice President, Nuclear
H.G. Stanley	Vice President, Nuclear Operations
G.T. Jones	Vice President, Nuclear Engineering
H. D. Woodeshick	Special Assistant to the President
E. W. Figard	Manager, Nuclear Information Services
T. C. Dalpiaz	Manager, Nuclear Maintenance
H. J. Palmer, Jr.	Manager, Nuclear Operations
C. A. Myers	Manager, Nuclear Regulatory Affairs
G. D. Miller	Manager, Nuclear Technology
G. J. Kuczynski	Manager, Nuclear Plant Services
J. V. Edwards	Manager, Nuclear Department Support
F. G. Butler	Manager, Nuclear Systems Engineering
A. F. Iorfida	Manager, Nuclear Procurement
T. R. Markowski	Dayshift Supervisor, Operations
J. M. Kenny	Licensing Group Supervisor
D. L. Hagan	Supervisor, Health Physics
R. R. Sgarro	Senior Project Engineer
T. Bannon	Project Engineer, Nuclear Licensing
R. R. Wehry	Compliance Engineer
R. L. Doty	Supervisor, Operations Technology

Others

David Ney	PA DER BRP, Nuclear Engineer
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ATTACHMENT 2

NRC SALP MANAGEMENT MEETING SLIDES

MAY 17, 1994



SUSQUEHANNA STEAM ELECTRIC STATION

SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE

MANAGEMENT MEETING
May 17, 1994

INTRODUCTORY REMARKS

W. Kane, NRC Deputy Regional Administrator

SALP PROCESS AND REPORT PRESENTATION

J. Durr, Acting Deputy Director, Division of Reactor
Projects

PP&L RESPONSE

DISCUSSION

CLOSING REMARKS

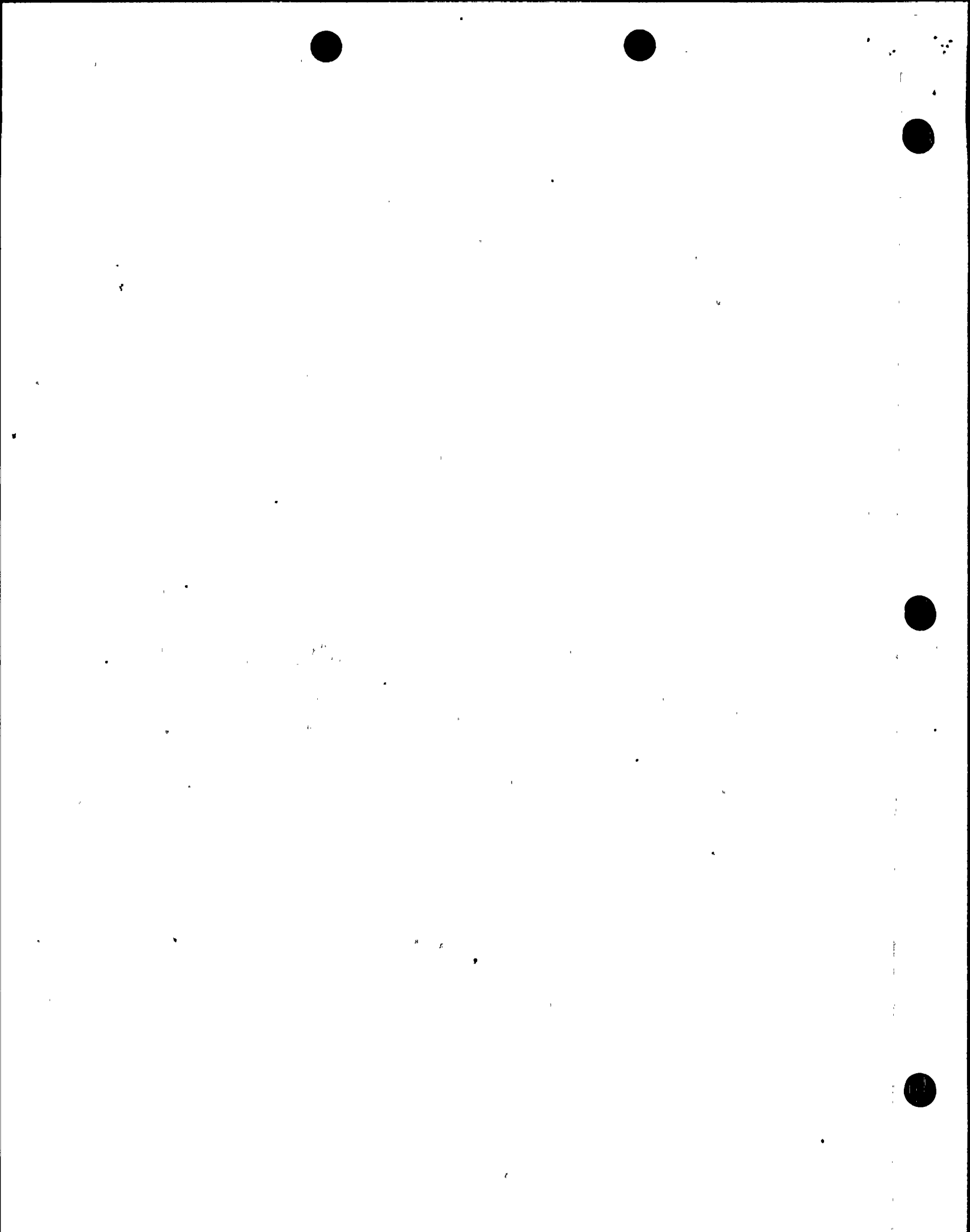
W. Kane



Susquehanna SALP
Management Meeting

Assessment Period
April 19, 1992 - February 26, 1994

U.S. Nuclear Regulatory Commission



Presentation

- Introduction W. Kane
- Report Presentation J. Durr
- Licensee Presentation PP&L
- Discussion
- Closing Remarks W. Kane

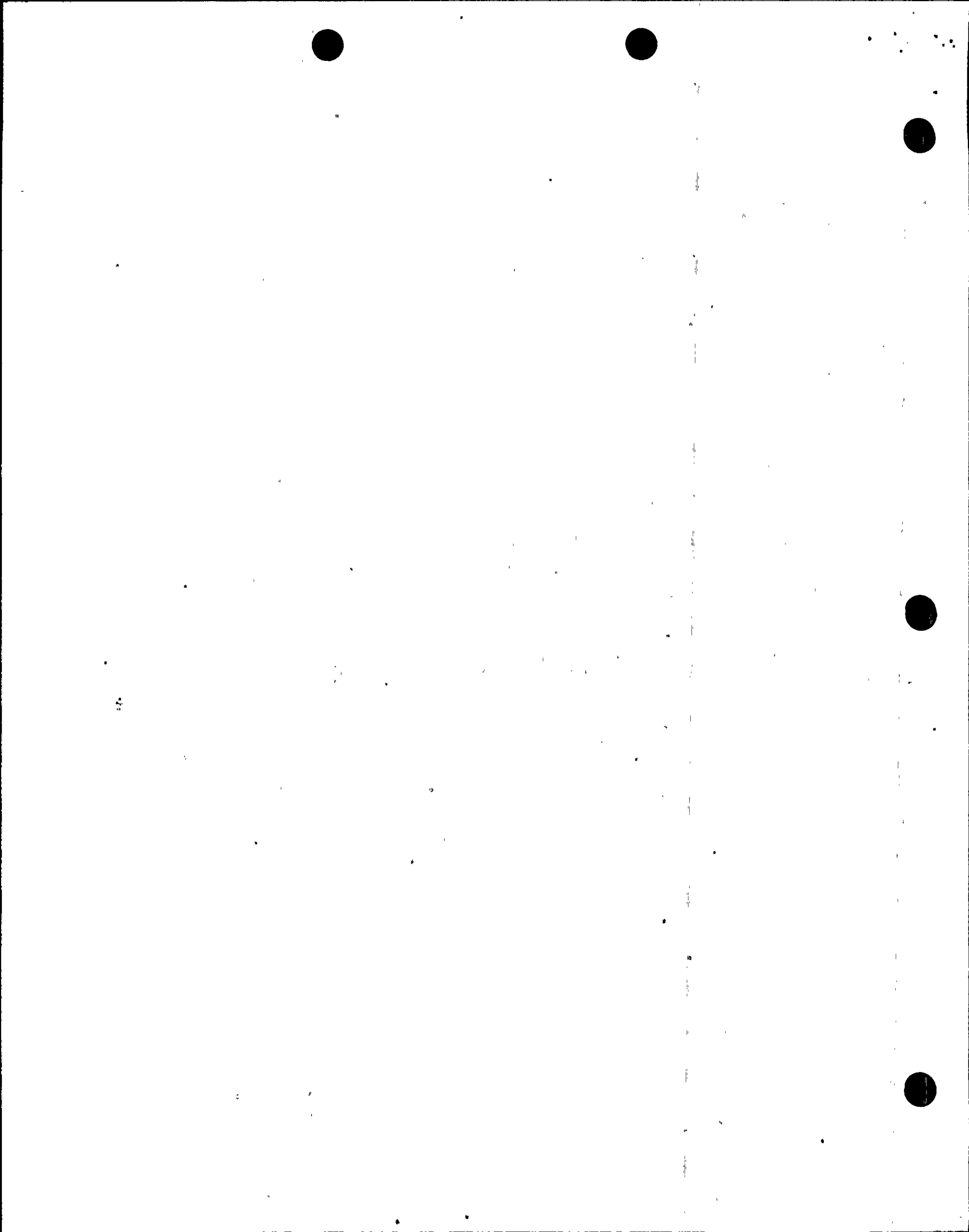
Revised SALP Process Effective July 14, 1993

- Changed from 7 functional areas to 4
- SA/QV incorporated into each area
- Emergency Preparedness, Radiation Protection, and Security combined into "Plant Support"
- SALP Board membership consists of 4 senior managers
- Emphasis on last 6 months of period
- Trends no longer included in category ratings



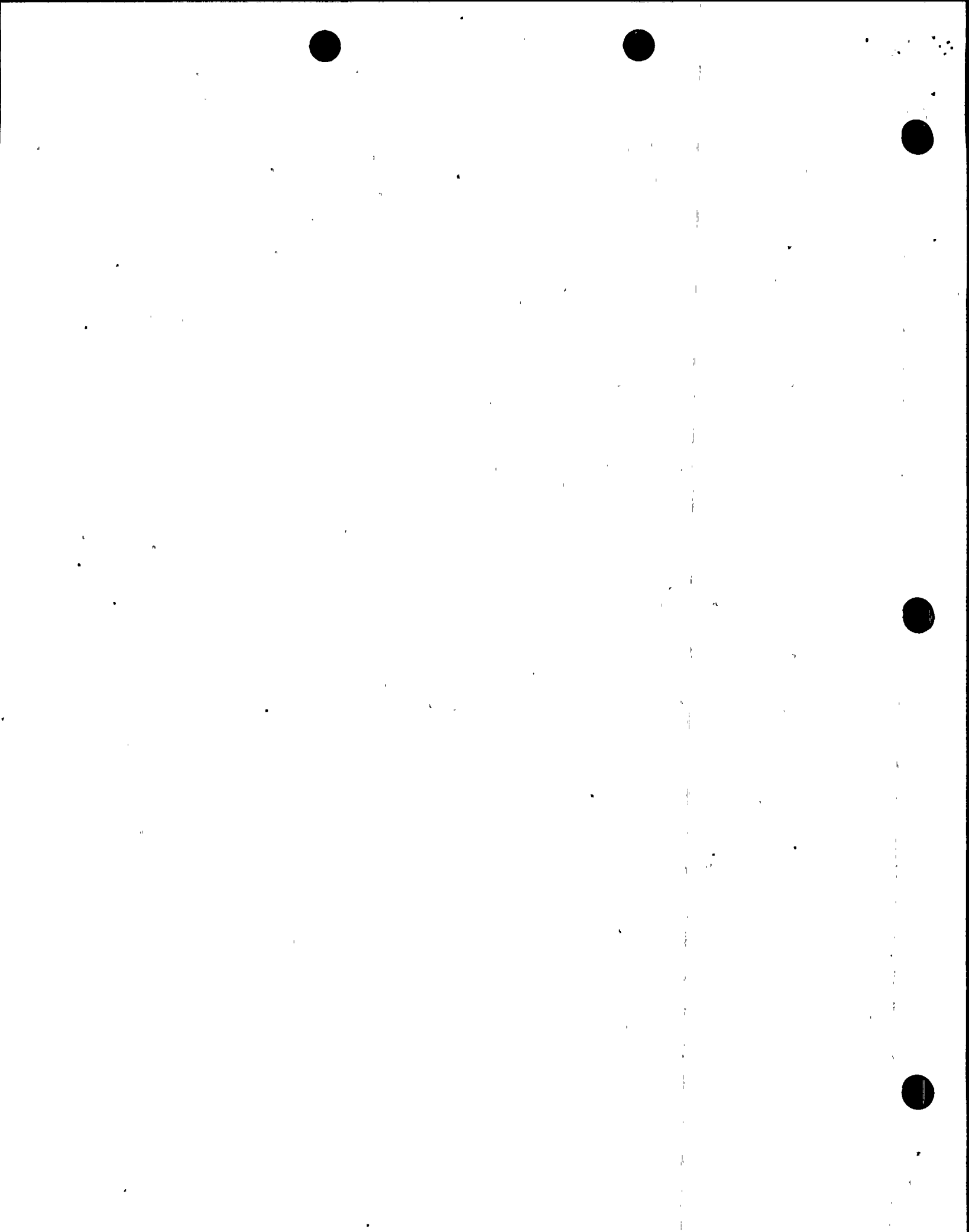
SALP Process Objectives

- Integrated Assessment
- Meaningful Dialogue
- Allocation of NRC Resources
- Inform Public



SALP Functional Areas

- Plant Operations
- Engineering
- Maintenance
- Plant Support
 - Radiation Protection
 - Emergency Preparedness
 - Security
 - Chemistry
 - Fire Protection
 - Housekeeping

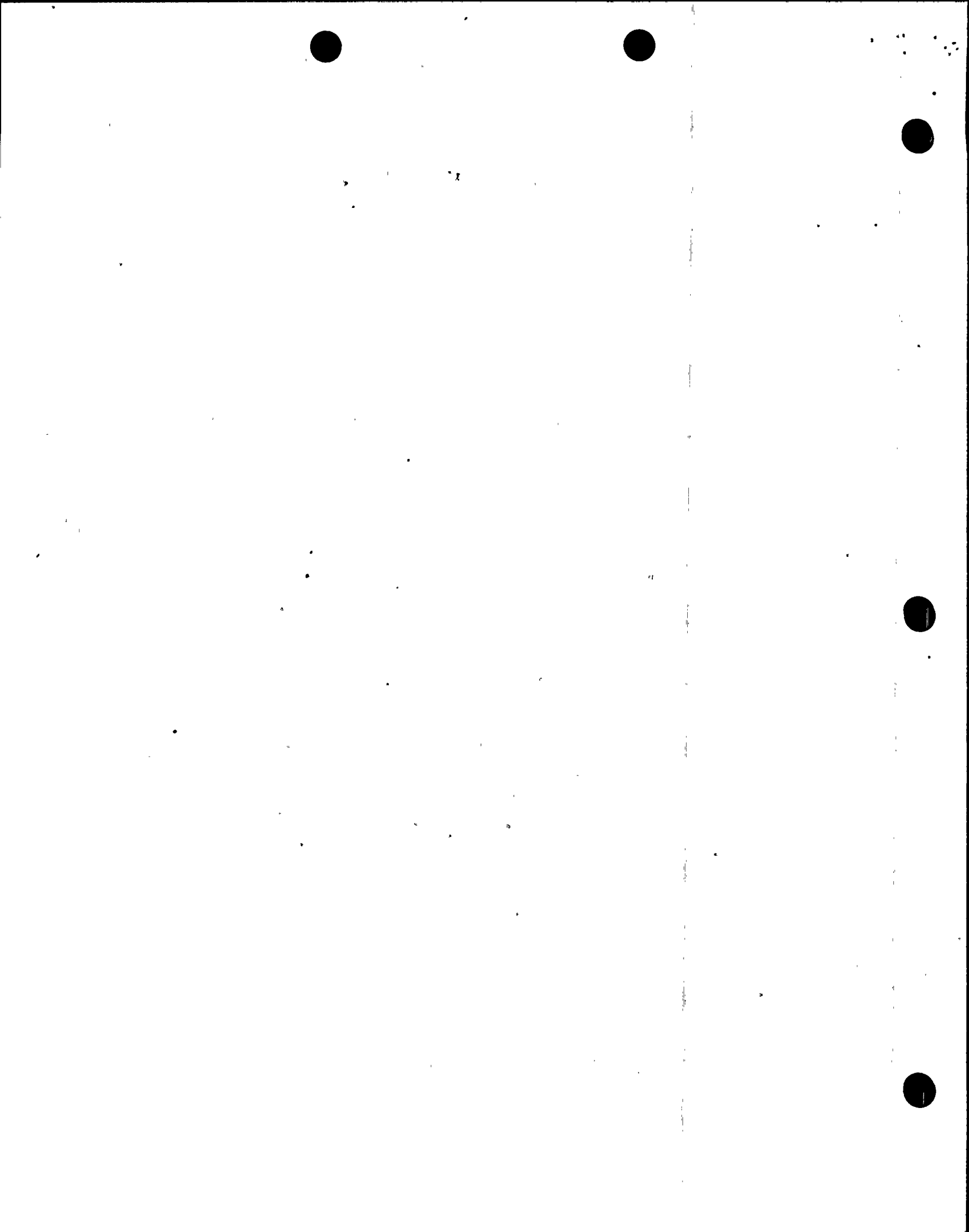


Performance Category Ratings

- Category 1 Superior Performance
- Category 2 Good Performance
- Category 3 Acceptable Performance

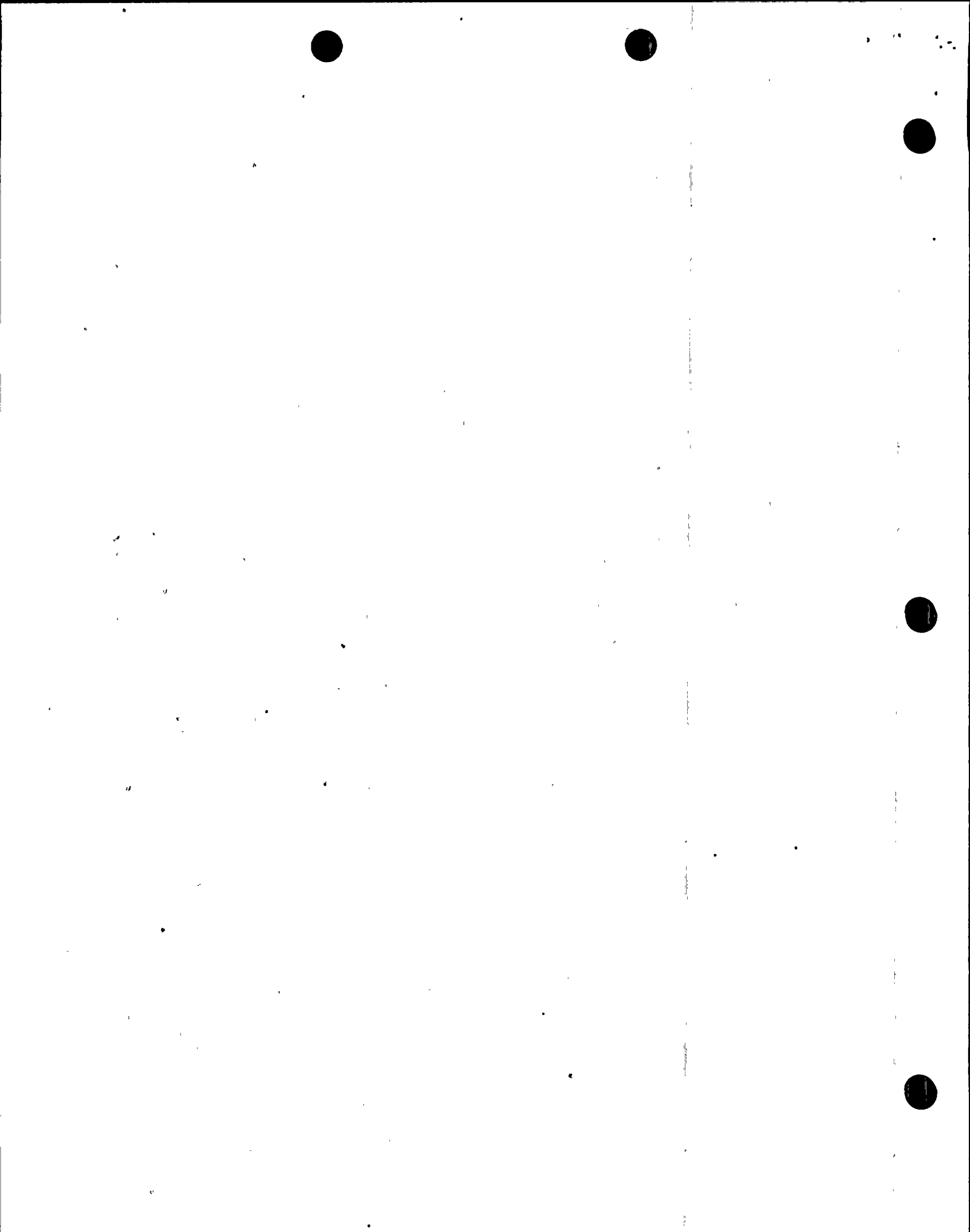
**SALP Category Ratings for the
Previous Period Ending
April 18, 1992**

■ Plant Operations	1, Declining
■ Maintenance	1
■ Engineering/Tech Support	1
■ Plant Support:	
Radiation Protection	2
Emergency Preparedness	1
Security	1
■ SA/QV	1



SALP Category Ratings for the Period Ending February 26, 1994

■ Operations	2
■ Maintenance	1
■ Engineering	1
■ Plant Support	1



Operations

Category 2

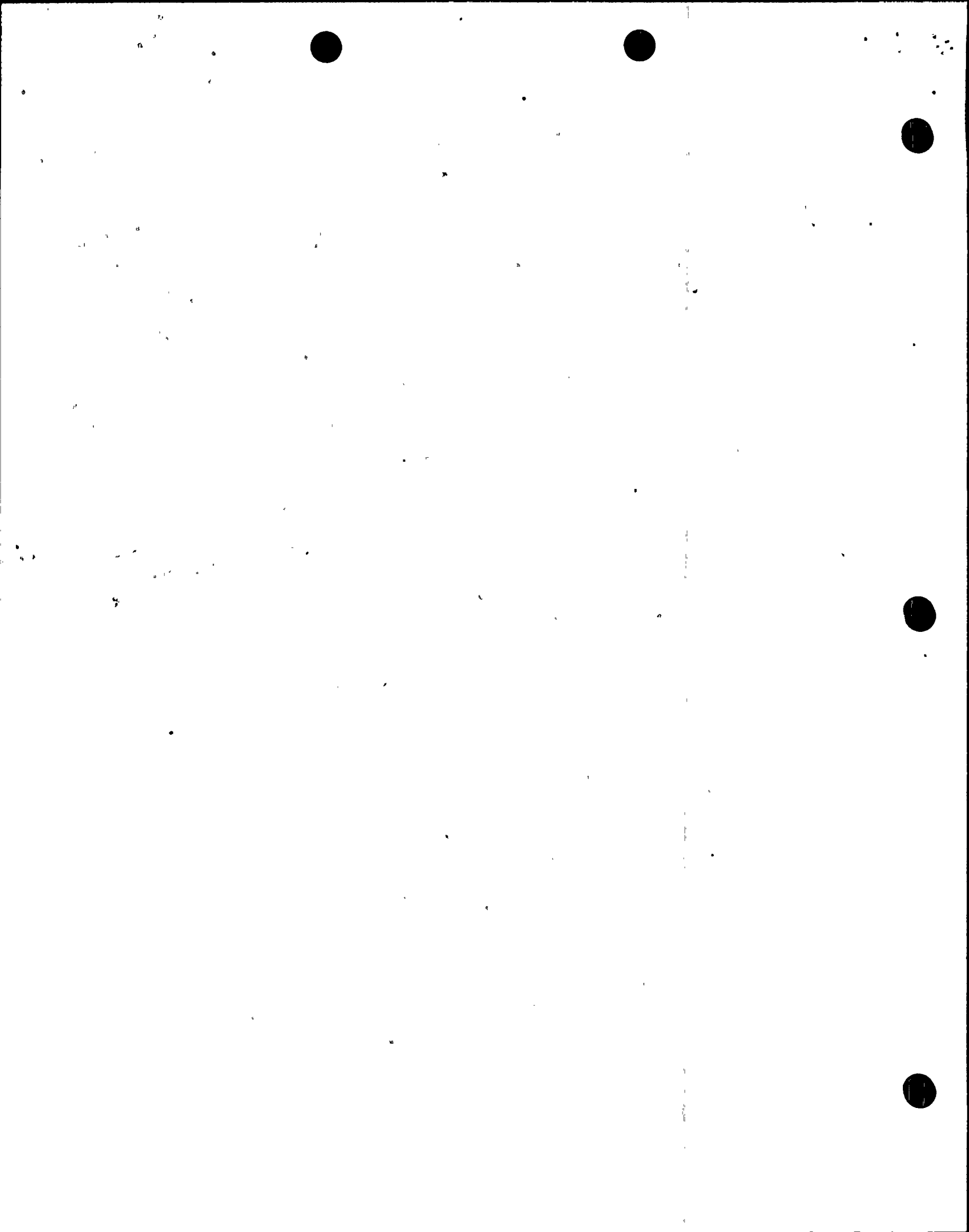
- Plant operational performance generally excellent
- Excellent management involvement in investigation of operational concerns
- Safety philosophy effectively conveyed to the operations staff
- Operations staff performance superior
- Operator training improved and effective
- Some aspects of performance diminished by recurrent problems



Maintenance

Category 1

- Maintenance program remained strong and well managed
- Excellent involvement of management
- Maintenance implementation was effective and strongly influenced by program and site management
- Exceptionally good plant material condition
- Excellent availability and reliability of safety systems
- High level of skill, experience and training in the maintenance staff
- Isolated instances of maintenance and surveillance activities below high standards



Engineering

Category 1

- Overall superior engineering with conservative safety perspective
- High quality engineering evaluations and plant modifications
- Good internal communications
- Comprehensive engineering training program
- Well managed system engineering program
- Effective engineering interface with site activities
- Very evident engineering management oversight and support of station activities
- Effectively managed EDR process
- Several repetitive events raised concerns over monitoring and trending system performance

Plant Support

Category 1

- Plant Support significantly contributed to safe plant performance
- Radiation Protection continued to improve
- Excellent Health Physics coverage
- Low stations doses
- Continued strong performance in Emergency Preparedness
- Important improvements implemented in preparation for the FFE
- Security program continued to be superior



Overview

- Excellent level of performance overall at SSES
- Effective management oversight; Management properly focused on safety issues
- Effective communications between all organizational units
- Continued strong performance of safety committees
- Effective performance feedback to management through self-assessments
- Generally excellent plant operational performance
- Well trained and knowledgeable personnel
- Personnel responded to plant events promptly and thoroughly
- Strong performance in Engineering and Maintenance
- Excellent Plant Support

ATTACHMENT 3

PP&L SALP MANAGEMENT MEETING SLIDES

MAY 17, 1994

SUSQUEHANNA OVERVIEW

H. G. Stanley
SSES SALP

May 17, 1994

COMMITMENT TO SAFE OPERATION

PP&L's nuclear safety culture is fundamental to our operation.

- Strong Safety Philosophy
- Training
- Personnel Safety
- Conservative Management Philosophy
- Critical Self Assessment
- Industry Involvement
- Deficiency Management

*H. G. Stanley
SSES SALP*

May 17, 1994



OPERATING PERFORMANCE

Susquehanna operations conducted in a safe, efficient manner.

- Consistent Attention to Industrial Safety
- Shutdown Risk Considerations Effectively Addressed via Outage Work Controls
- Decreased Number of Engineered Safety Feature Actuations
- Power Generation Remained High

*H. G. Stanley
SSES SALP*

May 17, 1994



ROOT CAUSE & CORRECTIVE ACTION

The prompt reporting and correction of problems remains a priority at Susquehanna.

- Lowered Threshold for Action to Evaluate & Solve Recurrent Problems
- Improving Investigations & Root Cause Analysis, Including Generic Implications
- Strengthened Ability to Probe Causes of Human Error
- Defined Expectations of Management Oversight
- Strengthening Assessment Process

H. G. Stanley
SSES SALP

May 17, 1994

SALP FUNCTIONAL AREAS

OPERATIONS
MAINTENANCE
ENGINEERING
PLANT SUPPORT

H. G. Stanley
SSES SALP

May 17, 1994

KEYS TO SUCCESS

- Quality Professionals
- High Standards
- Teamwork
- Training
- Continuous Improvement

H. G. Stanley.
SSES SALP

May 17, 1994

SUSQUEHANNA INITIATIVES

- System Trending
- Deficiency Management
- First Line Supervisor Oversight
- Procedural Adherence
- Plant Tooling & Equipment
- Maintenance Rule Implementation
- HP Benchmarking

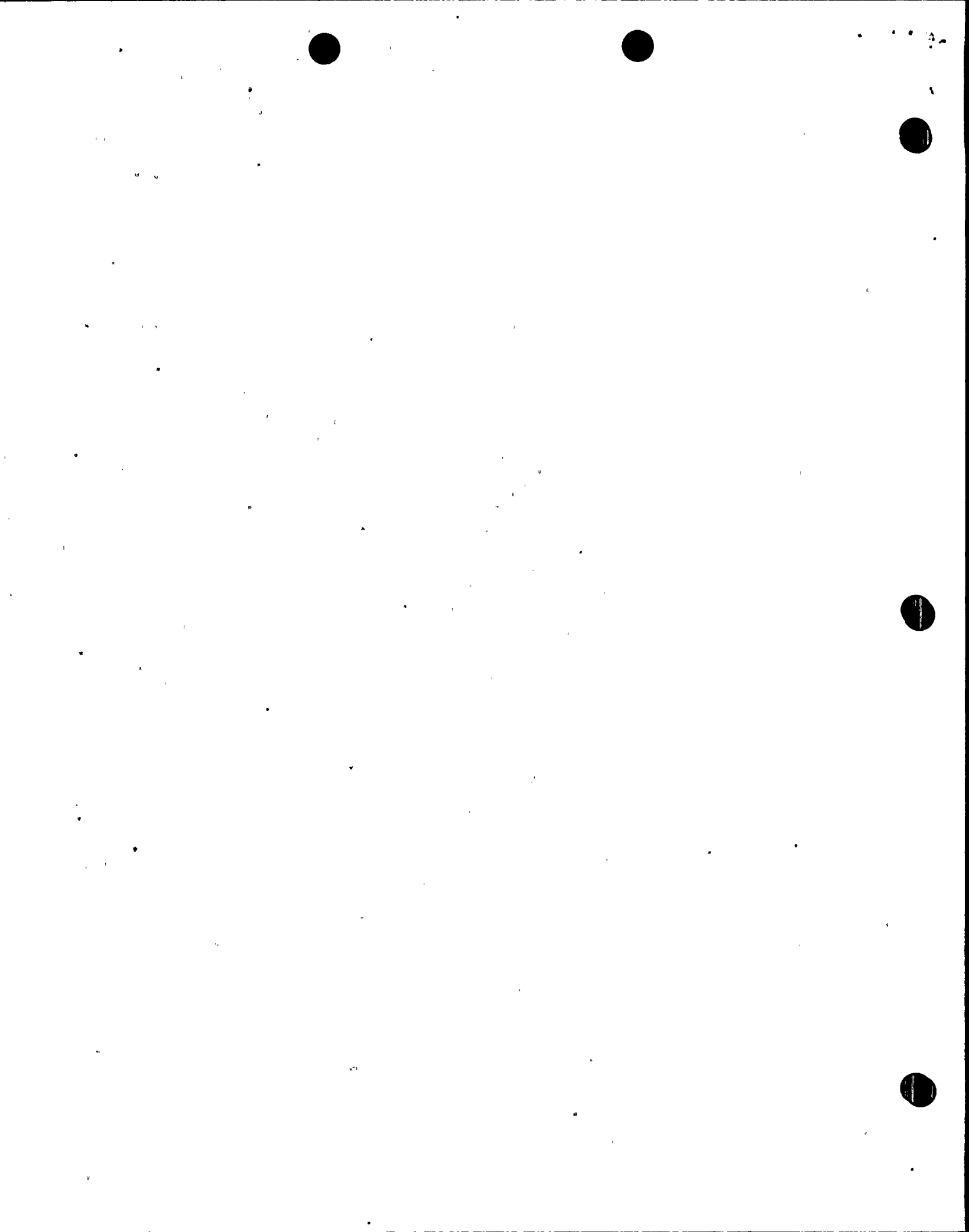
H. G. Stanley
SSES SALP

May 17, 1994

SUSQUEHANNA VISION

G. T. Jones
SSES SALP.

May 17, 1994



VISION

Our long-term VISION is to achieve excellence in the operation, maintenance, and support of Susquehanna.

- Safe Operation
- Continually Adjust Approach & Processes
 - Learn from Ourselves
 - Learn from Others

G. T. Jones
SSES SALP.

May 17, 1994

RECEIVED-REGION 1

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