U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Report Nos.	50-387/90-25; 50-388/90-25
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:	November 4, 1990 - December 29, 1990
Inspectors:	G. S. Barber, Senior Resident Inspector, SSES J. R. Stair, Resident Inspector, SSES S. M. Pindale, Resident Inspector, Salem

Approved By: P. Swetland 1/10/91

P. Swetland, Chief Date
Reactor Projects Section No. 2A,

<u>Inspection Summary:</u>

<u>Areas Inspected</u>: Routine inspections were conducted in the following areas: operations, radiological controls, maintenance/surveillance testing, emergency preparedness, security, engineering/technical support, safety assessment/quality verification, Licensee Event Reports, Significant Operating Occurrence Reports, and Open Item Followup.

<u>Results</u>: During this inspection period, the inspectors found that the licensee's activities were directed toward nuclear and radiation safety. No violations or deviations were identified. An Executive Summary is included and provides an overview of specific inspection findings.

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EXECUTIVE SUMMARY

Susquehanna Inspection Reports

50-387/90-25; 50-388/90-25

November 4, 1990 - December 29, 1990

Operations (71707, 71710, 60710, 71711)

Operators effectively controlled plant evolutions and identified plant problems.

A walkdown of the Emergency Service Water System identified only minor inconsistencies. Overall, the system was properly aligned and well maintained.

A review of refueling activities and startup preparations was performed which indicated that outage activities were appropriately conducted and controlled, and that the plant was ready for power operation.

Radiological Controls (71707)

Individual worker and Health Physics personnel implementation of radiological protection program requirements were observed with no noted inadequacies.

Maintenance/Surveillance (61726, 62703)

The licensee exercised good control of maintenance and surveillance activities.

A damaged piston pin bushing was discovered during a licensee inspection of the "D" emergency diesel generator. The licensee aggressively pursued this failure in an effort to determine the root cause. Further licensee action is needed to determine the specific root cause.

Emergency Preparedness (71707)

No emergency preparedness issues emerged during the period.

Security (71707)

Routine observation of protected area access and egress control showed good control by the licensee.

A degradation of the security data management system occurred during the period. This resulted from an undesirable practice of removing software from service during the backshift. However, compensatory measures were established in a reasonable time, thus limiting the consequences.

Engineering/Technical Support (71707, 92720)

The licensee submitted an update to a 10 CFR 50.9 report concerning the adequacy of the leak detection system for the main steam lines. Review of the adequacy of the safety evaluation is ongoing.

As a result of the licensee's deficiency reduction program, and reevaluation of previous reportability determinations, one issue was identified which should have been reported concerning control structure HVAC during a LOCA/LOOP. The inspector noted that the licensee took prompt compensatory action to ensure that the HVAC system would be capable of performing its safety functions.

Safety Assessment/Assurance of Quality(40500,92700,92701,92720)

A total of 98 Significant Operating Occurrence Reports were reviewed during the period.

Two licensee event reports were followed up in this report. Review of these indicated that appropriate reporting requirements were met, the events were adequalty reviewed, and corrective actions taken appeared adequate. Licensee efforts to enhance their event reportability criteria appeared to be benificial.

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 and 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 calculations, and selective review of applicable documents. Abbreviations are used throughout the text. Attachment 1 provides a listing of these abbreviations.

1.2 Susquehanna Unit 1 Summary

Unit 1 entered the inspection period in cold shutdown with refueling outage activities nearing completion. Startup was entered on November 13 and the reactor was made critical at 7:00 p.m. the same day. During heatup, the "A" reactor recirculation pump #1 seal did not stage properly. In addition, a "B" reactor recirculation system suction drain line valve incurred a packing steam leak. The unit was returned to cold shutdown on November 15, to correct these two problems. While shut down, the drain line valve packing was adjusted, but a decision was made not to replace the recirculation pump seal due to the likelihood that the seal would properly stage as pressure increased during startup. On November 16, startup recommenced. The main output breakers were closed on November 17, restoring Unit 1 power output to the grid.

On November 18, a steam leak on a main steam drain line was discovered and the unit was again shutdown to perform repairs. The unit was returned to service on November 19, following completion of repairs to the drain line. Elevated reactor coolant conductivity levels resulted in the licensee maintaining the unit at reduced power levels. On November 27, full power was restored following the identification and elimination of oil introduction into the feedwater system from the "B" reactor feedwater pump. Oil breakdown during passage through the reactor coolant system introduced byproducts into the system which resulted in the increased conductivity levels. Power ascension was resumed and full power was attained on November 27.

The plant remained at full power until December 2, when the unit was taken off line and returned to cold shutdown due to continuing problems with the "A" reactor recirculation pump #1 seal, higher than normal unidentified drywell leakage and decreasing level in the EHC oil reservoir. On December 6, the unit was restarted following replacement of the "A" Reactor recirculation pump #1 seal and the "B" reactor recirculation system drain line valves, and repairs to an EHC line. The unit was paralleled to the grid on December 7 and full power was reached on December 9.

On December 12, a service water leak in a pipe nipple from the iso-phase bus duct cooler occurred and power was run back to 70 percent to conduct repairs. Repairs were made to service water pipe, and power was restored to 100 percent on December 13. The unit was maintained at full power throughout the remainder of the period.

1.3 Susquehanna Unit 2 Summary

Unit 2 operated at or near full power for most of the inspection period. Scheduled power reductions were conducted during the period for control rod pattern adjustments, surveillance testing, and maintenance. On December 15, the unit was shutdown to commence a two day maintenance outage. Work performed during the outage included main generator ground testing, adjusting the torque switch setting on the RWCU F001 valve, and adding oil to the "B" recirculation pump motor reservoir. Startup commenced on December 16, and full power was restored on December 18. The unit remained at full power through the end of the inspection period.

2. OPERATIONS

2.1 Inspection Activities

The inspectors verified that the facility was operated safely and in conformance with regulatory requirements. Licensee management control was evaluated by direct observation of activities, tours of the facility, interviews and discussions with personnel. Safety system status, Limiting Conditions for Operation, and facility records were also reviewed. These inspection activities were conducted in accordance with NRC inspection procedure 71707.

The inspectors performed 222 hours of normal and back shift inspections including deep backshift inspections on: November 4, from 7:30 a.m. to 3:15 p.m.; November 9, from 3:00 a.m. to 6:00 a.m.; November 16, from 3:00 a.m. to 6:00 a.m.; December 7, from 3:00 a.m. to 6:00 a.m.; December 9, from 7:30 a.m. to 11:30 a.m.; and, December 16, from 7:30 a.m. to 1:30 p.m.

2.2 Inspection Findings and Review of Events

2.2.1 Engineered Safety Features Walkdown - Emergency Service Water System - Common

During the inspection period, the inspector independently verified the operability of the Emergency Service Water System (ESW) by performing a walkdown of the accessible portions of the system. The engineered safety system status verification included the following:

-- Confirmation that the licensee's system check-off lists and operating procedure were consistent with the plant as-built drawings and as-built configuration.

-- Identification of equipment conditions and items that might degrade performance.

-- Verification of proper breaker positions at local electrical panels and correct indications on control panels.

-- Verification that valves were in their proper positions, and that power was available.

-- Verification of good housekeeping in the area of the system equipment.

The following conditions were noted:

-- The 4160 Volt switchgear 1A20303 front enclosure was not properly secured since the top latching mechanism was not fastened. This appeared to be due to a bowing of the front enclosure door.

-- Some inconsistencies were noted in component descriptions between the system checklists and the labels on the components in the plant.

The inspector discussed the above observations with the licensee and was informed that action would be taken to correct the inconsistencies and to secure the switchgear enclosure door.

The inspector determined that the system was properly aligned in accordance with the operating procedure and the equipment conditions indicated that the components were generally well maintained. No further inadequacies were noted.

2.2.2 Review of Refueling Activities and Startup Preparations - Unit 1

2.2.2.1 Refueling Activities

The inspector attended daily plant meetings which reviewed and discussed ongoing refueling activities throughout the period. In addition, the inspector conducted numerous tours of the Unit 1 facilities to observe ongoing outage related work activities.

Major work activities completed during the outage included: removal and reload of core fuel bundles, replacement of 228 fuel bundles, 24 control rod drives, 20 LPRM strings, and 52 control rod blades. Other activities included numerous surveillances and inservice inspections. Special Projects completed included: heat exchanger replacements; containment radiation monitor replacements; conversion of the main turbine to partial arc operation; and upgrades of various valves, the emergency service water system, and the instrument air system. Major maintenance work completed included: inspection and cleaning of one RHR heat exchanger; inspection and repair of one natural draft cooling tower; inspections of the reactor vessel shroud support region, suppression pool liner plate coating, emergency core cooling system keepfill bypass, turbine generators, and reactor feedwater pump turbines. In addition, numerous other modifications were implemented.

Work activities observed by the inspector appeared to be conducted and controlled in accordance with appropriate procedures and applicable technical specifications.

2.2.2.2 Unit Startup Activities

Prior to startup following completion of the units fifth refueling outage, the inspector reviewed completed checklists for the RHR, RCIC, Core Spray, HPCI, SLC, and ESW systems. From this review, it appeared that the licensee had restored the systems to lineups required to support plant operation in accordance with applicable procedures and the units TS. In addition, the inspector performed an ESF walkdown of the ESW system. Results of this walkdown are discussed in Section 2.2.1 of this report. On November 16, the inspector observed portions of plant startup control room activities performed under GO-100-002, Plant Startup and Heatup, which included the withdrawal of control rods to establish criticality which was achieved at 5:19 a.m.. From these observations and reviews, it appeared that the licensee had adequately completed those steps necessary to provide assurance that the plant was operationally ready. No inadequacies were noted.

The licensee routinely prepares a Shutdown Action Item (AI) list for every planned or unplanned shutdown. The refueling outage AI list contained 15 items. All of these items were resolved prior to startup. No inadequacies were noted.

In addition to the above startup preparations, the inspector reviewed the licensee's Personnel Contamination Reports (PCRs) and ALARA evaluations and performance during the outage. This review is discussed in Section 3.0.

The licensee's progress in response to Generic Letter 89-10 concerning safety-related MOV operability and their In-Service Inspection (ISI) outage assessment was also reviewed by the inspector. This review is discussed in Section 4.4.2.

2.2.2.3 Plant Operations Review Committee (PORC)

The inspector observed licensee activities during startup PORC meetings conducted on November 7 and 8 to review major activities associated with the refueling outage. The PORC lasted three days starting on November 7 and the inspector observed activities on two of the three days. The PORC concluded that the plant was ready for startup after completion of certain action items. The items were completed and PORC subsequently recommended startup.

The inspector observed the PORCs review of a number of outage activities and concluded that the PORC's focus was on both worker radiation safety and nuclear safety. They challenged the staff on many issues and in many cases required followup. Followup actions were examined, in detail, and were not accepted unless found to be thorough and complete. No inadequacies were noted.

2.2.3 Incomplete Nuclear Instrument Surveillance

On November 29, the licensee requested a waiver of compliance for certain Nuclear Instrument (NI) surveillances since they could not be completed with the mode switch in the "Run" position. Certain rod blocks and scram functions for the SRMs, IRMs and APRMs require the mode switch to be in a position other than "Run". Their waiver request and subsequent amendment request were documented in PLA-3484 and PLA-3486, respectively.

The inspector questioned the licensee regarding certain aspects of the waiver. Specifically, the inspector was concerned with why these circumstances were not discovered earlier. The licensee stated that they believed that they were meeting the intent of the TS because they were performing the surveillance twice, once in Condition 1 and once in condition 2. They recognized that they could not accomplish the scram and rod block functions in Condition 1, and thus, wrote the surveillance procedure to bypass certain steps if the mode switch was in "Run". In addition, certain steps had to be "NA-ed" as directed by the procedure.

Notwithstanding the above, the licensee stated that although these circumstances existed since initial licensing, they were not questioned until an Instrument Technician questioned the practice of "NA-ing" the steps that could not be done with the mode switch in "Run". He questioned whether this practice met the intent of TS 3.3.1, 3.3.6 and 3.3.7.6. By "NA-ing" the steps, the licensee was able to sign off the surveillance as complete for the conditions that existed in Condition 1, when in fact a mode change was being made without completing the surveillance requirements. Further evaluation of this concern led the licensee to conclude that this practice was undesirable, and thus, sought relief in the form of a temporary waiver of compliance. The NRC reviewed the waiver request and asked the licensee to also provide an amendment request since NRC policy does not address a situation in which a needed shutdown is prevented. Thus, the amendment request was evaluated based on its own merits, independent of the waiver request. The amendment request was provided by the licensee on November 30. The NRC reviewed the proposed TS change, and denied it because it did not meet the requirements of 10 CFR 50.91 for exigent or emergency changes. These requirements do not address a condition that could prevent a plant shutdown, but only conditions that prevent startup or result in derating or shutdown. However, the NRC considered the safety significance of the requested change and found that strict compliance with TS 3.3.1, 3.3.6 and 3.3.7.6 in this case would not be in the interest of nuclear safety. Therefore by a letter dated November 30, 1990, NRC licensing documented that NRC would not take enforcement for not fully completing the required surveillance. However, this issue remains unresolved pending formal NRC review of the proposed TS change and NRC review of licensee activities that accepted the previous TS surveillances. (UNR 387/90-25-01)

3. RADIOLOGICAL CONTROLS

3.1 Inspection Activities

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 and 71711.

3.2 Inspection Findings

Personnel Contamination Reports

The licensee uses Personnel Contamination Reports (PCRs) to document skin or clothing contaminations. Poor or undesirable work practices accounted for 56% of all of the PCRs. Thirty-two individuals cross-contaminated themselves during work or while removing PCs, ten were caused by poor housekeeping, and another ten involved climbing or crawling in unsurveyed areas. Another significant contributor to the number of outage contaminations was clean area contaminations. Approximately 33% of outage contaminations occurred in areas not posted as contaminated. During this outage, nine were related to handling scaffold surveyed as clean and 37 were due to changing conditions including, spilling liquid, removing dry insulation and transferring contaminated items over boundaries. Although the total number of PCRs is not significant at this time, review of the causes indicates that significant improvement is achievable.

The numbers and types of PCRs were reviewed by PORC to determine if further reduction in the numbers of PCRs could be achieved. The licensee is implementing many improvements to reduce the number of PCRs. Some of them include additional training in the form of a videotape on the proper removal of PCs, upgrading rubber gloves and shoe covers, mandatory glove usage when handling scaffolding, and review of work area techniques to reduce contamination.

The inspector also noted that the licensee routinely used large area maslin swipes to survey clean areas and to survey the path taken by contaminated personnel on their way to be decontaminated. The inspector questioned the effectiveness of using large area maslin wipes for PCR surveys. The maslin swipes very rarely pick up loose surface contamination for these post-PCR investigative surveys. Smear surveys of these areas will often show levels of contamination just above the detectable threshold. The licensee needs to review this routine practice and assess its effectiveness.

The inspector observed radiological work practices during the outage and noted that they were generally good. However, control of loose tools and equipment at contaminated area boundaries could be improved. In one case, tools were noted to be loose across a step-off pad during HPCI work. In another case, a power cord was run across a contaminated boundary without being properly secured. HP was contacted and the conditions were corrected. Better housekeeping and awareness by work groups is needed.

ALARA

Licensee ALARA performance was a priority activity during the outage. With a majority of the outage work complete, approximately 270 man-rem was expended against a goal of 340 man-rem. Approximately, 78,000 RWP man-hours were expended. Eighteen man-rem was expended for ISI work against a goal of 34 man-rem. Nozzle inspections were completed significantly under the goal primarily due to using automated equipment. Previous efficiency improvements for snubber work were effective since early exposure totals were approximately 60% of the ALARA goal.

The inspector reviewed the licensee's ALARA performance and noted that eleven of eleven major work activities were completed within their radiation exposure goal. This good ALARA performance was primarily due to good planning and efficient work practices. However, some of this performance was due to the use of conservative estimates for area dose rates and work durations. A more accurate reflection of the radiation fields expected along with more exact work durations are needed to generate more challenging ALARA goals.

4. MAINTENANCE/SURVEILLANCE

4.1 Maintenance Inspection Activity and Observations

The inspector observed and/or reviewed selected maintenance activities to determine if work was conducted in accordance with approved procedures, regulatory guides, Technical Specifications, and industry codes or standards. The following items were considered, as applicable, during this review: Limiting Conditions for Operation were met while components or systems were removed from service; required administrative approvals were obtained prior to initiating the work; activities were accomplished using approved procedures and quality control hold points were established where required; functional testing was performed prior to declaring the involved component(s) operable; activities were accomplished by qualified personnel; radiological controls were implemented; fire protection controls were implemented; and the equipment was verified to be properly returned to service.

These observations and/or reviews included:

-- Eddy Current Testing of the 1E123B Turbine Building Closed Cooling Water Heat Exchanger on December 7.

-- Interim Inspection of the "A" Emergency Diesel Generator per WA S04013 on December 7.

-- Installation of a fabricated cover plate on the "C" Emergency Diesel Generator outboard bearing for use of the 250 Engine Analyzer per WA S05249 on December 14.

-- Inspection and Evaluation of the Broken 7R Piston Pin Bushing in the "D" Emergency

Diesel Generator on December 21.

4.2 Surveillance Inspection Activity and Observations

The inspector observed and/or reviewed the following surveillance tests to determine that the following criteria, if applicable to the specific test, were met: the test conformed to Technical Specification requirements; administrative approvals and tagouts were obtained before initiating the surveillance; testing was accomplished by qualified personnel in accordance with an approved procedure; test instrumentation was calibrated; Limiting Conditions for Operations were met; test data was accurate and complete; removal and restoration of the affected components was properly accomplished; test results met Technical Specification and procedural requirements; deficiencies noted were reviewed and appropriately resolved; and the surveillance was completed at the required frequency.

These observations and/or reviews included:

-- SI-178-210, Refueling Outage - Weekly Functional Test of Average Power Range Monitor Channels A - F, performed on November 9.

-- SI-178-226, Monthly Functional Tets of Rod Block Monitor Channels A & B, performed on November 9.

-- SO-100-011, Reactor Vessel Temperature and Pressure Recording, performed on November 16.

-- SO-156-007, Control Rod Coupling Check, performed on November 16.

4.3 Inspection Findings

The inspector reviewed the listed maintenance and surveillance 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 surveillance and 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. No unacceptable conditions were identified, but one work activity required followup. Two other activities were reviewed during a startup PORC. The details are provided below.

4.3.1 Cracked "D" Diesel Generator Piston Pin Bushing

On December 19, the licensee identified a cracked piston pin bushing in the "D" diesel generator (DG) during a 25-hour inspection. The cracked bushing was found during a visual examination of the "7R" cylinder. In addition, small pieces of the bushing were found in the sump. Further licensee investigation showed no discoloration of other piston pins or

bushings. There was no other damage noted on either the cylinder liner or the piston itself. The licensee is reviewing inspection results from the other diesels to reconfirm their acceptability. This piston and bushing had been previously changed out in response to sandblast grit in the turbocharger intercooler (see Inspection Report 50-387/90-20). The connecting rod was not changed out in response to this sandblast entrainment.

The piston was removed and disassembled on December 20. The licensee's metallurgist examined the bushing and concluded that the material was that specified and that there were indications of burnt oil in the fractured areas. No material related problems were identified. The licensee also examined their installation procedures and found that they were very comprehensive.

The inspector examined the bushing and fragments and discussed the issue with the licensee and the vendor. The inspector questioned the licensee on their installation procedures and noted that the licensee blue checks each piston pin for 80% contact area in addition to performing a comprehensive set of dimensional checks. The inspector also questioned the vender representative (Cooper rep) and noted that Cooper presses the bushing into the piston as a part of its refurbishment procedure. Numerous dimensional checks and a thorough cleaning is also part of the refurbishment. The Cooper rep postulated that the failure was due to foreign material entrainment between the piston and bushing that resulted in a stress riser and cracking in a localized area. Loading the diesel exacerbated the problem and led to the cracking found. The licensee and the vendor believe this failure was an isolated case. An NRC technical representative reviewed the failure onsite and at the licensee's analysis laboratory. This item is unresolved pending resolution of the failure mechanism and its potential generic consequences. (UNR 387/90-25-02)

4.3.2 Safety-related MOV Operability

Generic letter (GL) 89-10 imposed additional requirements to ensure the operability of safety-related (SR) MOVs. This GL required licensee's to review and reverify their present design for all SR MOVs. Design reverification requires checking the following SR MOVs attributes: torque switch setting, limit switch setting, adequate grease, motor nameplate data, and spring pack performance. Also, MOVs needed to close during line breaks events were to be reviewed with the highest priority.

The licensee committed to review 399 SR MOVs for GL 89-10 purposes by the end of 1994. Their schedule requires 302 MOVs to be reviewed during refueling outages and 97 MOVs to be done during non-outage periods. The licensee is in the process of implementing a new valve testing methodology (VOTES) with a much improved accuracy and repeatability. This methodology uses strain gages on the valve's yoke to determine the actual thrust developed during valve closure. Twenty-four SR MOVs were tested during the outage with good results. One SR MOV had interferences that prevented completing testing before the end of the outage. All tested SR MOVs have been reviewed with torque switch settings found to be in their specified range.

The inspector observed the VOTES testing methodology and discussed it with the licensee. Although the number of SR MOVs tested was low, the licensee gained experience with the new VOTES methodology. Correlation between VOTES and the previous testing method was generally good. All exceptions were appropriately dispositioned. The NRC will continue to monitor licensee implementation of MOV testing upon issue of detailed inspection guidance.

4.3.3 In-Service Inspection

The licensee performed 3832 In-Service inspections (ISI) during the outage. Some were new inspections required by the ISI program, while others were required to monitor previous indications to comply with NCR corrective action. The number due to NCRs has dropped steadily. The number of ISI-related NCRs has decreased from 305 NCRs to 90 NCRs between the Unit 1 Third Refueling outage and the current outage. Four old NCRs were generated during previous RPV Internal Visual examinations. NCR 90-0235 documented cracks in four shroud head bolts which were subsequently replaced in this outage. NCRs 90-0242 and 90-0251 documented cracks in the tie rod capture plate and the dryer hood areas which were dispositioned "use-as-is " since the cracks were in a low stress area, with no wet steam; and showed no crack growth. NCR 90-0250 documented cracks in the dryer support ring and was dispositioned "use-as-is" since there is no concern for loose parts.

The inspector reviewed the licensee's assessment of these NCRs. The first three were adequately dispositioned and monitored. NCR 90-0250 documents cracks discovered four outages ago in the dryer support ring which have grown 79, 32, 50 and 35 mils in each successive outage. Per the vendor, this crack growth is expected. Currently, the most limiting crack has grown to approximately 0.410" or 35% of the allowable growth (1.54") necessary to maintain structure integrity. A critical crack depth is specified and licensee efforts will be focused on generating appropriate repair/replacement activities prior to reaching this critical depth.

5. EMERGENCY PREPAREDNESS

5.1 Inspection Activity

The inspector reviewed licensee event notifications and reporting requirements for events that could have required entry into the emergency plan.

5.2 Inspection Findings

No events were identified that required emergency plan entry. No inadequacies were identified.

6. SECURITY

6.1 Inspection Activity

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.

6.2 Inspection Findings

The inspector reviewed access and egress controls throughout the period. No unacceptable conditions were noted.

6.2.1 Security System Degradation

The Security Data Management System (SDMS) was degraded at 10:08 p.m., December 7. A portion of the SDMS was experiencing problems restarting. As a result, both the primary and alternate SDMS were halted. Prior to this halting, compensatory measures were established. After the SDMS failed to promptly reboot following the halt, the licensee strengthened the existing compensatory measures by recalling additional security officers. A major portion of the SDMS was restored to a normal status 38 minutes later and compensatory measures were subsequently secured. The NRC was notified in accordance with 10 CFR 73.71.

The inspector reviewed the specific details of the event with the licensee on December 9 and noted that certain practices aggrevated the event. First, the practice of halting both the primary and alternate SDMS on backshift hours was undesirable because of the limited number of software support personnel available. Second, although compensatory measures were acceptable, increased staffing levels are normally available on dayshift. Planning the halting evolution on dayshift would have improved the licensee's ability to provide more comprehensive compensatory measures. Third, the SDMS's design could be enhanced to make it less susceptible to these types of failures by modification to the hardware. The licensee has agreed to review each of the aforementioned items. Their response will be reviewed in future inspections. This is an unresolved item. (UNR 387/90-25-03)

7. ENGINEERING/TECHNICAL SUPPORT

7.1 Inspection Activity

The inspector periodically reviewed engineering and technical support activities during this inspection period. The on-site Technical (Tech) section, along with Nuclear Plant Engineering (NPE) in Allentown, provided engineering resolution for problems during the inspection period. The Tech section generally addressed the short term resolution of problems while NPE scheduled modifications and design changes, as appropriate, to provide long lasting problem correction. The inspector verified that problem resolutions were thorough and focused at preventing recurrences. In addition, the inspector reviewed short term actions to ensure that the licensee's actions provided reasonable assurance that safe operation could be maintained.

7.2 Inspection Findings

7.2.1 10 CFR 50.9 Report Update - Leak Detection System Design Basis Reconstitution

The licensee provided an update to a 10 CFR 50.9 report they originally submitted on the leak detection system (LDS) for the main steam lines. The original report was supplemented as a result of the licensee's efforts to reconstitute the LDS's design basis. The LDS's function is to detect relatively small steam leaks and to isolate the affected penetration if the leak is of sufficient magnitude to warrant isolation. The original LDS design basis required the timely detection and isolation of leaks sized at 25 gpm or greater. The licensee's current analysis shows that the system will alarm for a leak sized at a 25 gpm leak and isolate a 50 gpm leak with the current Technical Specification setpoints. The ability to meet the original design basis is still being evaluated by the licensee and questioned by the NRC.

The NRC has conducted numerous conference calls with the licensee on the leak detection system's design basis and the NRC has questioned the licensee in detail. The current concern is an outgrowth of NRC reviews of this issue and resulted from the licensee finding that the original HVAC calculations were inadequate in predicting the actual temperature profiles that would result from a steam leak in various spaces in either the Unit 1 or Unit 2 Reactor Buildings. The inspector is continuing to review the adequacy of the licensee's safety evaluation and the adequacy of the licensee's leak detection systems design basis reconstitution effort.

7.2.2 Overall Deficiency Reduction Program - Inability to Maintain the Control Structure Pressurized During a LOCA/LOOP

As part of a comprehensive program to reduce the number and impact of open deficiencies, the licensee is reviewing all open Nonconformance Reports (NCRs), Significant Operating Occurrence Reports (SOORs), and Engineering Discrepancy Reports (EDRs). This is being done as a part of the licensee's overall deficiency reduction program. In addition to assessing significance, basis for continued operation, and the adequacy of schedules for closure of these deficiencies; a re-evaluation of the previous reportability determinations was performed using current philosophy and NRC guidance. The licensee recently lowered their reporting threshold by emphasizing the need to evaluate reportability based on adverse safety consequences for the uncorrected discovered conditions. The following item was deemed to meet thresholds such that had it occurred recently, it would have been determined reportable.

NCR 87-0279 documented a the failure of four suction dampers in the Control Structure

HVAC system to remain open for a postulated LOCA/LOOP scenario. These dampers would, in fact, fail closed. They are normally maintained open via non-safety related Instrument Air supply which is assumed to be unavailable for this accident scenario. The dampers failing closed would prevent maintaining the required positive 1/8 inch water gauge pressure in the Control Structure.

This condition was originally identified on July 2, 1987 and the licensee subsequently reported it based on applying the new reporting criteria for these issues. At the time this event was identified, PORC recommended that the subject dampers be wired open to ensure the proper alignment during a LOCA/LOOP. This was performed per bypass 1-87-045. Also of concern was the effect of this action on Chlorine isolation of the system since the dampers were required to close on High Chlorine. Closure of these specific four dampers was not needed to prevent chlorine introduction to the control structure since positive closure of a different set of four outside air supply dampers ensured the safety function. The licensee was concerned that a failure of the four suction dampers for the control structure heating and ventilation fans (OV103A/B) and the computer room floor cooling fans (OV115A/B) could result in a slightly negative pressure in the control room. To ameliorate this concern, procedures were modified to trip fans OV103A&B and OV115A&B when chlorine was detected. These compensatory actions were performed to ensure that the HVAC system was capable of performing its safety functions.

The inspector reviewed the licensee's actions for this event and noted that the licensee took prompt compensatory action to correct the nonconforming condition when it occurred. The licensee determined, in 1987, that the event was not reportable since the compensatory action corrected the non-conforming condition. The inspector had no further questions at this time.

8. SAFETY ASSESSMENT/QUALITY VERIFICATION

8.1 Licensee Event Reports (LER), Significant Operating Occurrence Report (SOORs), and Open Item (OI) Followup

8.1.1 Licensee Event Reports

The inspector reviewed the following LERs:

Unit 1

90-016-00 Equipment Exceeded Qualified Life per Environmental Qualification Program. This event was reviewed in special NRC Inspection Report 50-387/90-17.

90-017-01 Followup to LER 90-017. Secondary Containment Isolation Division I Automatic and Manual Initiation Functions Lost Due to a Relay Failure. This event was reviewed in NRC Inspection Report 50-387/90-15.

*90-019-00 LCO 3.0.3. Entries Due to No Containment Radiation Monitor Aligned to Drywell.

*90-020-00 "As Found" Main Steam Line Penetration Leakage Rate Exceeds Technical Specification Limits.

The inspector verified for each listed LER that the details of the event were clearly reported, including the accuracy of the description of the cause and the thoroughness of corrective action. The inspector determined whether further information was required from the licensee, whether generic implications were involved, and whether the event warranted onsite followup. The events that required on-site followup are denoted by asterisks and are reviewed in more detail in Section 8.1.1.1. No unacceptable conditions were identified for the LERs not requiring on-site followup.

8.1.1.1 On-site Followup of Licensee Event Reports

For those LERs selected for on-site followup (denoted by asterisks in Detail 8.1.1), the inspector verified that the reporting requirements of 10 CFR 50.73 had been met, that appropriate corrective action had been taken, that the event was adequately reviewed by the licensee, and that continued operations of the facility was conducted in accordance with Technical Specification limits. The following findings relate to the LERs reviewed on site:

LER 90-019-00, L.C.O. 3.0.3 Entries Due to No Containment Radiation Monitor (CRM) Aligned to Drywell

On September 9, the licensee discovered that on three occasions between September 6 and 9, the "B" CRM System was removed from service to allow the "B" H_2O_2 analyzer to be swapped from the drywell to the suppression chamber for oxygen sampling while the "A" CRM system was inoperable. This resulted in T.S. L.C.O. 3.0.3 being entered as a result of having no CRM in service aligned to the drywell. During the first occurrence it was not recognized by personnel that the L.C.O. was, in effect, entered due to the failure to record the fact that the "A" CRM was inoperable on shift turnover logs. On the following two occasions, the L.C.O. was intentionally entered to allow performance of the required daily suppression chamber oxygen sampling.

Licensee review of these occurrences resulted in the determination that personnel involved did not fully appreciate the need to exhaust all other alternatives prior to entering L.C.O. 3.0.3 and that they entered the L.C.O. in order to adhere to the stations policy of procedural compliance.

Corrective action included repairing the "A" CRM, restoring it to operable status on September 9, and training of operations pesonnel on both the importance of accurate log-keeping and the importance of exhausting all viable alternatives prior to taking actions which would place the unit in LCO 3.0.3. In addition, procedural revisions reflecting station policy with regard to entering LCO 3.0.3 were implemented which require prior approval of the Supervisor of Operations. This is to assure appropriate management approval and consideration of appropriate compensatory actions.

The inspector reviewed the LER and discussed the event with appropriate Licensee personnel. Although licensee action in response to this event was found to be appropriate, there is a need to reemphasize that the NRC considers entry into LCO 3.0.3 to be on a very limited basis and that in order to do so requires senior management involvement and comprehensive compensatory actions. Failure to do so in this case led to inappropriately entering LCO 3.0.3.

90-020-000, "As Found" Main Steam Line Penetration Leakage Rate Exceeds TS Limits

On September 20, the licensee determined that the "as found" leakage from the main steam line (MSL) penetration local leak rate tests (LLRTs) was in excess of the limit allowed by Technical Specification 3.6.1.2(c). Specifically, the minimum as found MSL containment penetration leakage rate was 110.9 scfh. This required leakage rate is less than 46.0 scfh. The unit was in its fifth refueling outage at the time of the tests. The licensee made the appropriate notification to the NRC on September 20, within the required time.

The "C" MSL inboard and outboard main steam isolation valves which were the major contributors to the leakage (99.6 scfh) were reworked and a post-maintenance LLRT was performed. Total MSL containment penetration leakage was reduced to 20.9 scfh. These valves are manufactured by Atwood and Morrill Company, Inc. and are Model No. 21190-H. Licensee inspection of the "C" MSL MSIVs revealed only slight areas of light oxidation and minor surface scratching, neither of which were considered contributors to the cause of the high "as found" leakage rate. An assessment of the potential safety consequences was performed and it was determined that no safety significance or risk to the health and safety of the public was incurred due to the "as found" leakage rate being within the capacity of the MSIV-leakage control system.

The inspector reviewed the LER and discussed the event with appropriate licensee personnel. It should be noted that the inability to determine the exact cause of the high "as found" leakage is an industry generic problem which the BWR Owners group and the NRC have been evaluating for appropriate resolution. The inspector was informed that a Topical Report from the BWR Owners Group was submitted to the NRC in December 1990. The inspector had no further questions concerning this topic.

8.1.2 Significant Operating Occurrence Reports

SOORs are licensee reports that provide prompt problem identification and tracking, short and long term corrective actions, and reportability evaluations. The licensee uses SOORs to document and bring to closure problems identified that may not warrant an LER.

The inspectors reviewed the following SOORs during the period to ascertain whether: additional followup inspection effort or other NRC response was warranted; corrective action discussed in the licensee's report appears appropriate; generic issues are assessed; and, prompt notification was made, if required:

Unit 1

73 SOORs inclusive of 1-90-350 through 1-90-426

Unit 2

25 SOORs inclusive of 2-90-142 through 2-90-166

8.1.3 Open Items

8.1.3.1 Updated) UNR 50-388/89-05-01: Determine Generic Applicability of the Failure of the Solenoid Valve for Reactor Building Chilled Water System Valve No. HV-28792B2

The solenoid valve was functionally tested and disassembled onsite by the valve vendor (ASCO). Although a positive failure mode was not determined, the vendor concluded that the valve may have malfunctioned because of moisture induced corrosion or other debris that may have impeded movement of internal critical valve parts; some rust-like debris was found within the valve internals during disassembly.

The licensee's Nuclear Design organization investigated the failure and developed specific recommendations and conclusions, which were forwarded to station personnel. The recommendations and conclusions also considered the results of the vendor's investigation.

The inspector reviewed the related documents and interviewed the responsible station engineering personnel. The inspector found that multiple factors may have contributed to the valve failure. Specifically, 1) the setting of the filter regulator on the gas supply (20-25 psi) was at the low end of the valve's pressure range (15 - 150 psi), 2) the operating temperature of the valve was at the upper end of the normal operational range for the valve (150 degrees F), 3) the failure occurred toward the end of the qualified life for the solenoid valve's elastomers, and 4) a low level of physical contamination was present.

The licensee's review of this event discounted several of the recommendations when evaluated separately. However, the inspector concluded that the above contributing factors, when reviewed and evaluated collectively, may warrant additional action and/or may determine that the specific application of the solenoid valve may not be optimal. Additionally, a previous failure of a similar solenoid valve also occurred in which the cause was not positively identified. The inspector therefore concluded that additional review of the recommendations and conclusions is necessary for full resolution of this issue. The fact that this issue was not resolved for almost 2 years was indicative of the licensee's discrepancy management program. Further attention to this matter is needed. This item remains open.

9. MANAGEMENT AND EXIT MEETINGS

9.1 Routine 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.

9.2 Attendance at Management Meetings Conducted By Region Based Inspectors

Dates(s)	Subject	Inspection Report No.	Reporting Inspector
11/16	Water Chemistry and ISI	90-19; 90-19	H. Kaplan and R. Harris
11/29	Security	90-23; 90-23	G. Smith
11/16	Engineering	90-24; 90-24	D. Moy and R. Mathews

9.3 Reportability Meeting

NRC managers and inspectors visited the licensee's Allentown office on November 15 to further review the licensee's event reporting philosophy especially as it relates to design basis issues. During the meeting, the licensee discussed new reportability criteria being implemented to lower their reporting threshold for these types of events. There was a good exchange of ideas between the NRC and the licensee. Attachment 2 is the handout that was provided during this Reportability meeting.

ATTACHMENT 1

Abbreviation List

AD	- Administrative Procedure
ADS	- Automatic Depressurization System
ANSI	- American Nuclear Standards Institute
CAC	- Containment Atmosphere Control
CFR	- Code of Federal Regulations
CREOASS	- Control Room Emergency Outside Air Supply System
CRM	- Containment Radiation Monitor
DG	- Diesel Generator
DX	- Direct Expansion
ECCS	- Emergency Core Cooling System
EDR	- Engineering Discrepancy Report
EP	- Emergency Preparedness
EPA	- Electrical Protection Assembly
ERT	- Event Review Team
ESF	- Engineered Safety Features
ESW	- Emergency Service Water
EWR	- Engineering Work Request
FO	- Fuel Oil
FSAR	- Final Safety Analysis Report
ILRT	- Integrated Leak Rate Test
JIO	- Justifications for Interim Operation
LCO	- Limiting Condition for Operation
LER	- Licensee Event Report
LLRT	- Local Leak Rate Test
LOCA	- Loss of Coolant Accident
LOOP	- Loss of Offsite Power
MOV	- Motor Operated Valve
NCR	- Non Conformance Report
NDI	- Nuclear Department Instruction
NPE	- Nuclear Plant Engineering
NPO	- Nuclear Plant Operator
NRC	- Nuclear Regulatory Commission
OI	- Open Item
PC	- Protective Clothing
PCIS	- Primary Containment Isolation System
PMR	- Plant Modification Request
QA	- Quality Assurance
RCIC	- Reactor Core Isolation Cooling
RG	- Regulatory Guide
RHR	- Residual Heat Removal
RHRSW	- Residual Heat Removal Service Water
RPS	- Reactor Protection System

RWCU	- Reactor Water Cleanup
SGTS	- Standby Gas Treatment System
SI	- Surveillance Procedure, Instrumentation and Control
SO	- Surveillance Procedure, Operations
SOOR	- Significant Operating Occurrence Report
SPING	- Sample Particulate, Iodine, and Noble Gas
TS	- Technical Specifications
TSC	- Technical Support Center
WA	- Work Authorization

Reportability
Meeting with NRC

Objective of Meeting:

To reach agreement on how PP&L should proceed with reporting in the interim until new NRC guidance is issued.

Agenda

- Definition of safety significance.
- Recent PP&L letter. Basis for PP&L position.
- NRC perspective.
- Review of past reportability determinations.
    - Control Structure Chilled Water
    - MSIV Closure
    - Delta-T concerns
    - Other issues
- Improvements/Changes to PP&L process.

<u>*Safety Significance*</u>

All discrepancies will be initially screened for safety significance using the following questions as guidance.

- Does the discrepancy appear to adversely impact a system or component explicitly listed in the Tech Specs?

- Does the discrepancy appear to compromise the capability of a system or component to perform as described in the FSAR?

- Does the discrepancy appear to adversely impact any applicable licensing commitments?

If the answer to any of the above questions is yes, PP&L will expeditiously initiate operability and reportability evaluations.

Determining Safety Significance

PP&L Direction

- EDRs and SOORs receive thorough review for safety significance.
- NCRs have not received centralized review for safety significance.
- All discrepancy management programs will involve a thorough review of issues for safety significance using formal established criteria.

Process

- Deficiencies will be identified, documented, and evaluated for safety significance.
- Operability will be promptly determined and appropriate compensatory actions taken.
- Timely action will be taken to resolve issue regardless of operability, with emphasis on safety significance of issue.
- Safety significance will consider consequence of failure rather than potential for failure.
- Conclusions will be thoroughly documented.
- Reportability will proceed independently from operability determinations.

Interim Reporting Process

- PP&L will focus on safety. Our threshold for 50.72/50.73 will be lowered.
- We will take prompt corrective action.
- A dedicated team will preside over an improved EDR process.
- Issues reported under 50.9 will receive same level of attention as issues reported per 50.72/50.73.
- Issues may begin as 50.9 and evolve into LERs.
- NCRs will get more consistent and extensive review for reportability.
- When in doubt, PP&L will review issues with Senior Resident to obtain feedback on appropriate reporting mechanism.

Reportability Determinations

Subject	How Reported	New Criteria
Emerg. Switchgear Room Cooling	50.9	50.72/50.73 USQ: Consequences of Malfunction
MSIV Closure	Originally Not Reported Voluntary LER	50.72/50.73 USQ: Probability of Malfunction
Steam Leak Detection	50.9	50.9: Significance does not meet USQ threshold

Emergency Switchgear Room Cooling

Description

During a DBA, cooling for Unit 1 emergency switchgear room is provided by Control Structure Chilled Water instead of Reactor Building Chilled Water.

A new single failure mechanism was discovered that could preclude proper operation of CSCW for ESWGR room cooling following a DBA.

Original Reportability Evaluation

Based on the ability to detect, analyze, and react to this postulated single failure, it was concluded that no immediate operability issue existed and that the issue was not reportable under 50.72/50.73. A 50.9 report was made.

Reevaluation Using USQ Criteria

- Increase the probability of occurrence or consequences of an accident or malfunction evaluated in FSAR?
  - Yes; subsequently determined that ability to detect and react was inadequate.
- Create the possibility for a different type of accident?
  - Yes; results in potential loss of multiple safety systems.
- Reduce the margin of safety as defined in the basis for Tech Specs?
  - Yes; potential loss of multiple safety systems reduces margin of safety for long-term cooling function.

Reportable per 50.72/50.73

MSIV Closure

Description

Based on NRC Information Notice 88-51, PP&L investigated and determined that the inboard MSIVs would not fully close without pneumatic-assist under design containment pressure conditions. The FSAR indicates that MSIVs will close with pneumatic and, or spring force.

Original Reportability Evaluation

Evaluation showed that under DBA conditions, actual containment pressures would be lower than design. As a result, the valves would actually shut with either pneumatic or spring force.

Reevaluation Using USQ Criteria

- Increase the probability of occurrence or consequences of an accident or malfunction evaluated in FSAR?
  - Yes; loss of diversity in component design is an apparent increase in probability.
- Create the possibility for a different type of accident?
  - No.
- Reduce the margin of safety as defined in the basis for Tech Specs?
  - No.

Reportable per 50.72/50.73.

Steam Leak Detection

Description

As a result of continuing reviews resulting from Reactor Building temperature concerns, numerous steam leak detection issues were raised:

- Room isolations would occur with 25gpm leaks rather than 5gpm leaks as stated in the FSAR.
- Backdraft isolation dampers would isolate too early to permit actuation of system isolation due to steam leak. (Later found to be acceptable)
- Leaks in Main Steam Tunnel can be masked by coolers.

Original Reportability Evaluation

Early evaluations were extremely conservative and overstated the potential consequences. As better information was developed, the consequences were determined to be minimal. Due to the number of issues and uncertainty, reporting under 50.9 was appropriate.

Reevaluation Using USQ Criteria

- Increase the probability of occurrence or consequences of an accident or malfunction evaluated in FSAR?
  - No.
- Create the possibility for a different type of accident?
  - No.
- Reduce the margin of safety as defined in the basis for Tech Specs?
  - No.

Not reportable per 50.72/50.73

Reportability Determinations

Subject	How Reported	New Criteria
Electrical Distribution	Originally 50.9 Later reported as LER	50.72/50.73: USQ: Consequences of malfunction
Degraded Grid	50.9	50.9: Significance does not meet USQ threshold
Dynamic Quals Racked-Out Breakers	50.9	50.9: Significance does not meet USQ threshold

Electrical Distribution

Description

In preparation for the EDSFI, several electrical concerns were raised. Specific single failures were found to affect load shedding following certain events. This had the potential to reduce voltages in one case and to overload diesels in another case.

Original Reportability Evaluation

The first concern was evaluated to result in higher loads on a 480V MCC. However, calculated voltages were still acceptable. The second concern could result in higher loads on the diesels following an event. Additional calculations were needed to assess the impacts of these loads. This was reported to NRC per 50.9 on 5/25/90.

Subsequent Reportability Evaluation

On 7/20/90, PP&L concluded that failure of a 125V DC battery channel coincident with or just prior to a LOCA and a LOOP could result in overloading of an emergency diesel. This independent failure had to occur shortly before the LOCA/LOOP since loss of the battery channel would result in a 2-hour LCO with hot shutdown required within 12 hours. If the failure occurred a fraction of a second after the LOCA/LOOP, load shedding would proceed normally. This was reported to NRC per 50.72/50.73.

Reportability Determinations

Subject	How Reported	New Criteria
Reactor Building Temperature	50.9	50.72/50.73: Consequences of Malfunction
Diesel Inlet Air Temperature	50.9 (Original diesel failures were 50.72/50.73)	50.9: PP&L does not believe that this represents the cause of the diesel failures.
Control of Heavy Loads	50.9	50.9: Significance does not meet USQ threshold

Reportability Determinations

Subject	How Reported	New Criteria
Degraded Grid Issues	50.9	50.9: Significance does not meet USQ threshold
Limitorque MOVs	Not reported	Not reportable.
Reactor Water Cleanup F001	Originally not reported Later reported as LER	50.72/50.73: PP&L was confused by prior NRC letter

Conclusions

- Discrepancy management process is rapidly evolving.
- Improvements will be made in processing of discrepancy documents.
- Improvements will be made in documenting decisions and their bases.
- We will focus on safety significance.
- We will lower our threshold for 50.72/50.73.
- We will achieve closure of discrepancies within one cycle.

ENCLOSURE 2

AGENDA

PP&L MEETING WITH NRC
DISCREPANCY MANAGEMENT

Management Perspective	*Al Male*
Engineering Discrepancy Management	*Glenn Miller*
Department Action	*George Kuczynski*
Department Direction	*Gene Stanley*

MANAGEMENT PERSPECTIVE

PP&L has a strong record regarding the resolution of issues related to design and operation.

- *We take aggressive measures where safety is challenged.*

- *Our people do high quality technical work.*

- *Our management is involved.*

MANAGEMENT PERSPECTIVE

We have implemented major changes to enhance our management of all discrepancies:

- *Established a process for managing engineering discrepancies.*
- *Established new standards for closure.*
- *Accelerated closure activities.*
- *Assessed our actions against past events.*
- *Realigned engineering.*

MANAGEMENT PERSPECTIVE

In late 1989, external inputs initiated PP&L's ongoing self assessment process regarding discrepancy management:

- *Design-related allegations*
- *Questioning of our threshold for reporting*

We were responsive to those inputs:

- *Engineering Discrepancy Program initiated*
- *Reporting of safety-significant emerging design issues*

MANAGEMENT PERSPECTIVE

Since that time, PP&L has focused on the basic objectives of discrepancy management in order to properly assess our effectiveness:

- ***Early identification and verification***
- ***Prompt operability determinations, including compensatory actions***
- ***Prompt reportability determinations***
- ***Prioritization of corrective actions based on safety significance***
- ***Timely closure of all discrepancies***
- ***Proper documentation of decisions***
- ***Feedback to originator***

MANAGEMENT PERSPECTIVE

Engineering and plant staff have worked together and achieved these objectives during several efforts in 1990.

- *Diesel events - With the team in place, our corrective action plans and interactions with NRC met our expectations.*

- *Bypasses - The backlog has been reduced significantly.*

- *EDSFI - The NRC had positive comments on our integrated response team.*

- *Discrepancy Review Committee - Integrated assessment of engineering discrepancies.*

- *NCR Closure - Significant progress has occurred.*

MANAGEMENT PERSPECTIVE

In mid 1990, PP&L enacted an action plan to improve our effectiveness in meeting our defined objectives. Near-term activities which have been or are nearing completion are:

- *Communicate new standards and the need for a change in our performance in closing discrepancies*
- *Establish clear guidance on operability and reportability determinations*
- *Institute program improvements*
- *Review the backlog of EDR's, SOOR's, and NCR's**
- *Test the adequacy of our actions through root cause analysis*

* *EDR: Engineering Discrepancy Report*
SOOR: Significant Operating Occurrence Report
NCR: Non-Conformance Report

NEW STANDARDS

Nuclear Department Policy Letter 90-003 clearly communicates management's expectations.

We must:

- *Provide an open environment to identify and report discrepancies*
- *Ensure proper attention and priority*
- *Ensure visibility of discrepancy closure*
- *Ensure that action plans for closure are in place*
- *Assume responsibility for assessing our attitude and effectiveness*
- *Ensure that the lifetime of a discrepancy is generally no more than one cycle of operation*

OPERABILITY/REPORTABILITY

Timely determinations regarding operability and reportability are an integral part of the discrepancy management process.

- *PP&L supports the NRC's efforts to provide guidance.*
- *PP&L has and will continue to meet regulatory requirements.*
- *PP&L also recognizes that we must sometimes look beyond the letter to the spirit of the law in order to ensure that the NRC gets the information they need to fulfill their mission.*
- *We believe that we are in fundamental agreement.*
- *Our programs have been strengthened to improve the documented basis and the timing.*

PROGRAM IMPROVEMENTS

We have implemented actions that will support our discrepancy management objectives.

- *Engineering Discrepancy Program*
  - *Discrepancy Review Committee*
  - *Root Cause Analysis*
- *Training*
- *Integrated Management Approach*

BACKLOG REVIEW

We have conducted a review of old items; no safety significant discrepancies have been identified.

Status:

- ***NCR, EDR reviews complete***
- ***SOOR review ongoing***

We will not allow a backlog of old issues to reaccumulate.

ROOT CAUSE ANALYSIS

Objective: To ensure that our discrepancy management action plan was comprehensive.

Process: We evaluated our effectiveness in resolving four specific engineering issues by examining how they were handled from their initiation to the present.

Results: Our current direction was confirmed.

ROOT CAUSE ANALYSIS

The results of the study noted several strengths.

- *The engineering organization has a proven high level of technical competence.*
- *Engineering strives to achieve high standards (perfection) in the conduct of their business.*
- *The Engineering Discrepancy Program facilitates identification of issues.*
- *When our organization focuses on an issue, we drive it to closure.*

ROOT CAUSE ANALYSIS

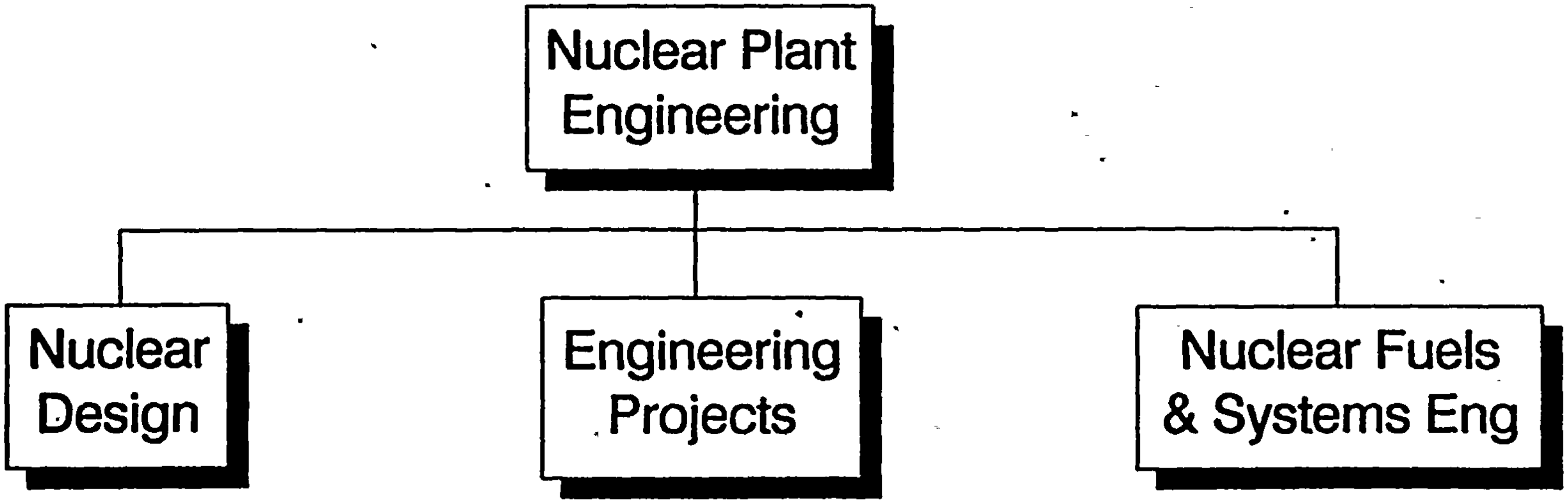
The results confirmed that the actions we had implemented were the right ones. The analysis said we needed:

- ***An integrated process.***
- ***Design Basis Documentation.***
- ***Operability and reportability guidance.***
- ***Management focus on closure.***

MANAGEMENT PERSPECTIVE

To achieve the desired results we need:

- *Management focus.*
- *Clear accountability for results.*
- *Committed resources.*
- *Grouping of related functions.*


Nuclear Plant Engineering
Nuclear Design
Engineering Projects
Nuclear Fuels & Systems Eng

ENGINEERING DISCREPANCY MANAGEMENT

Discrepancy Management Direction - *The current process has achieved its objective of identifying engineering issues. This process is improved by introducing other essential elements of discrepancy management.*

- *Accountabilities for pursuit of closure are focused organizationally.*
- *Process includes validation and screening for items of safety significance.*
- *Operability and reportability determinations are integral to the process and are strengthened.*
- *High departmental visibility to issue closure will ensure success.*
- *Design Basis Documentation improvements will provide a stronger foundation in the future.*

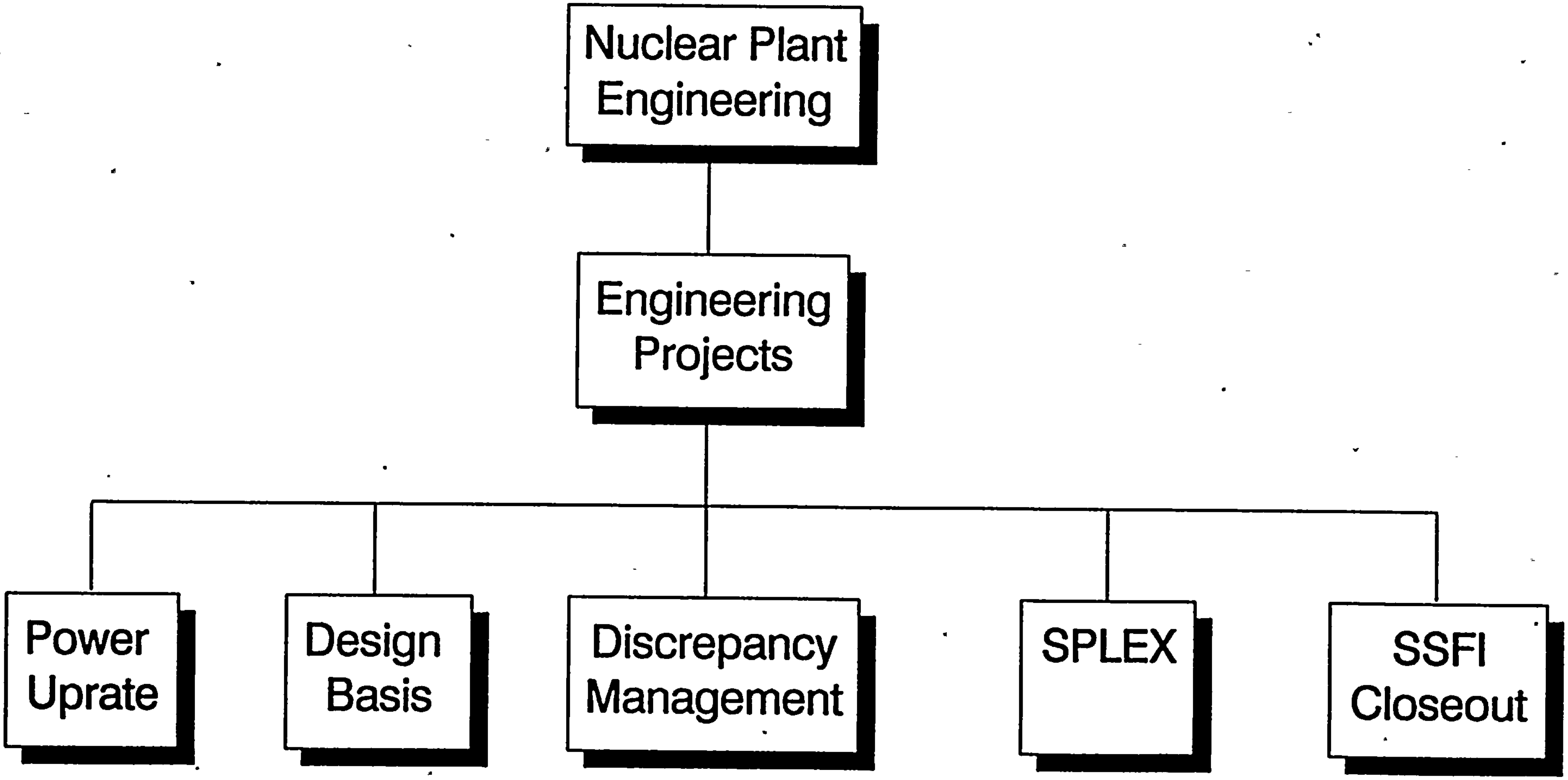
ENGINEERING DISCREPANCY MANAGEMENT

Engineering Organization - The realigned organization both facilitates and focuses efforts key to resolution of engineering discrepancies.

- *Dedicated management to oversee all aspects of discrepancy issues.*

- *Dedicated engineering resources to pursue resolution of discrepancies.*

- *Design Basis Documentation project both aids in resolution of issues and provides foundation for future.*


Nuclear Plant Engineering
Engineering Projects
Power Uprate
Design Basis
Discrepancy Management
SPLEX
SSFI Closeout

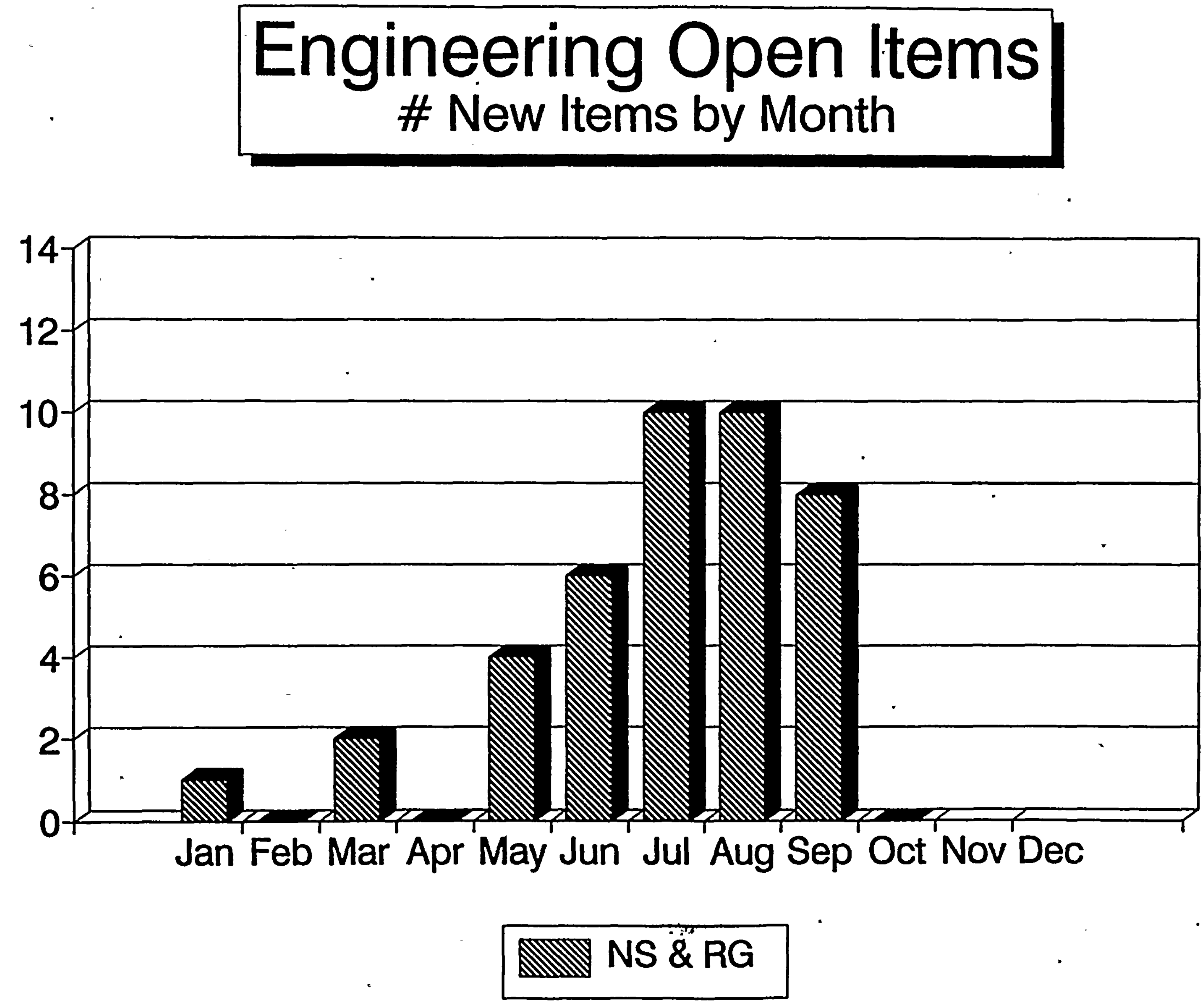
ENGINEERING DISCREPANCY MANAGEMENT

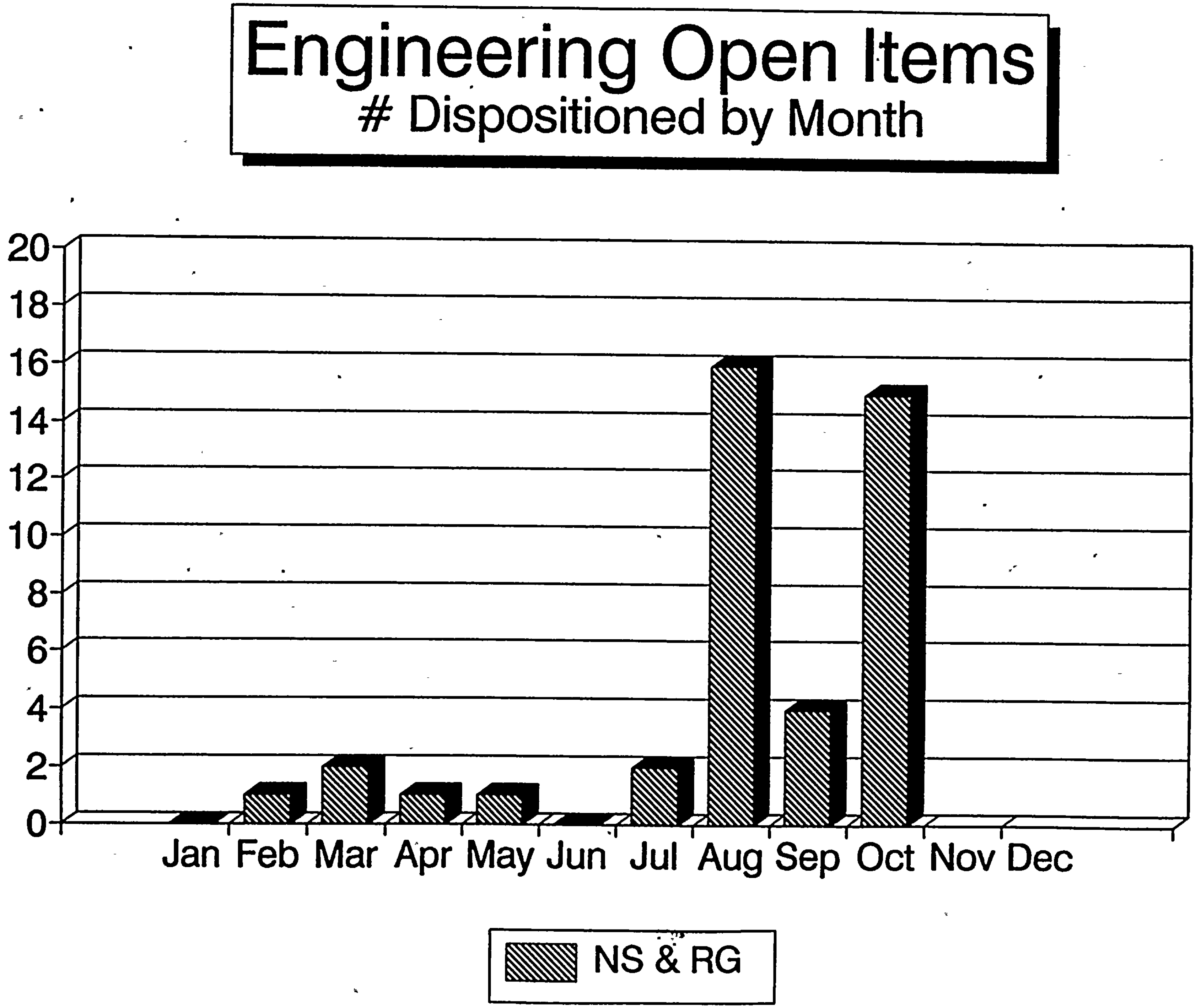
Current Status - *The backlog of open issues was reviewed. This backlog is characterized as follows:*

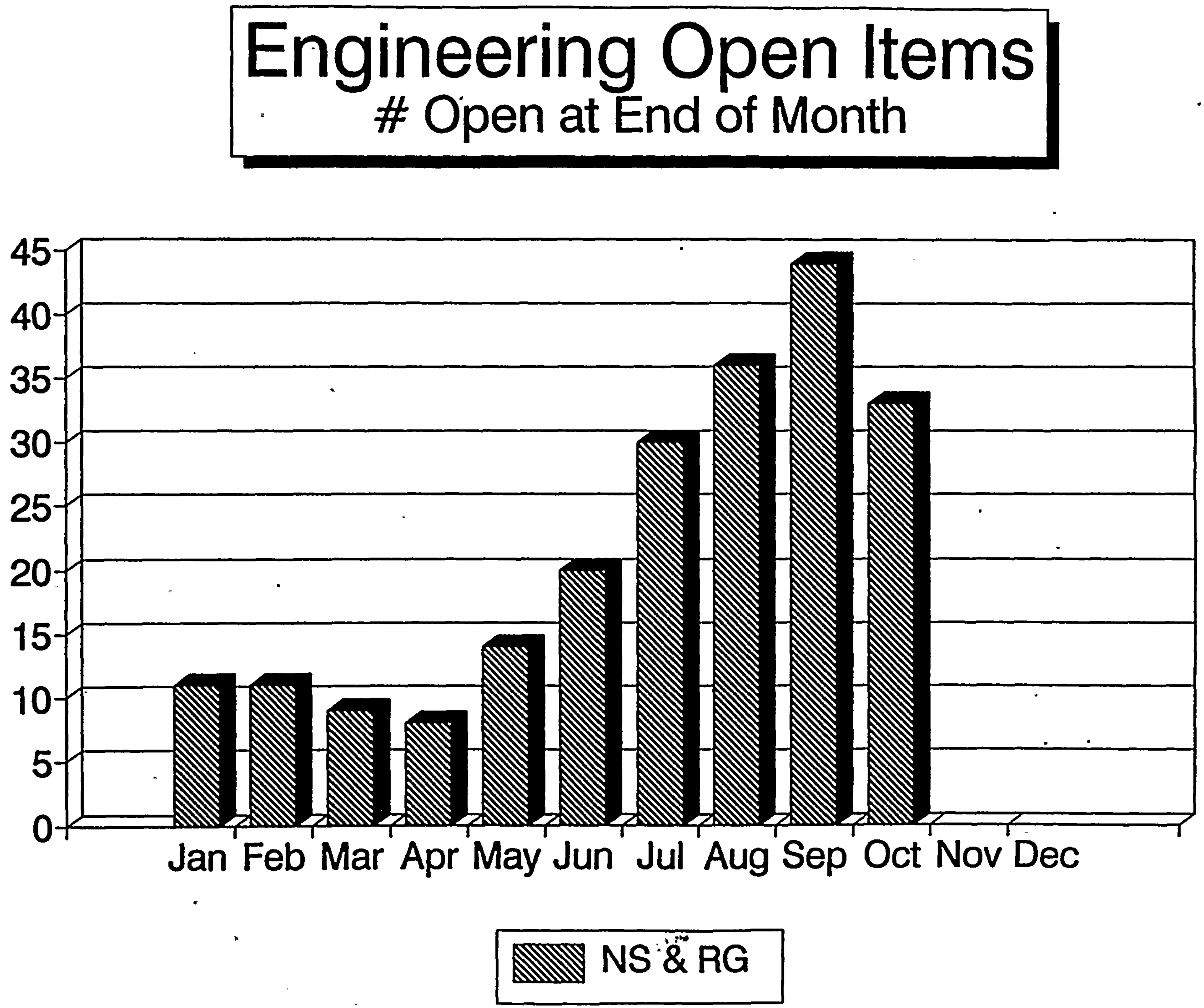
- *There are 33 items classified as "Nuclear Safety" or "Regulatory", all of which are dispositioned. Schedules for closure are either in place or under development.*

- *There are 172 items classified as "Technical," "Management", or "Economic". None of these items is a significant engineering deficiency.*

- *Plans are being developed for the closure of open items.*


Engineering Open Items
New Items by Month
14
12
10
8
6
4
2
0
Jan Feb Mar Apr May Jun Jul Aug Sep Oct Nov Dec
NS & RG


Engineering Open Items
Dispositioned by Month
20
18
16
14
12
10
8
6
4
2
0
Jan
Feb
Mar
Apr
May
Jun
Jul
Aug
Sep
Oct
Nov
Dec
NS & RG


Engineering Open Items
Open at End of Month
45
40
35
30
25
20
15
10
5
0
Jan
Feb
Mar
Apr
May
Jun
Jul
Aug
Sep
Oct
Nov
Dec
NS & RG

ENGINEERING DISCREPANCY MANAGEMENT

EDSFI Project - The EDSFI closeout project serves as a model for the way we intend to resolve discrepancies in the future.

- *Significant resources are dedicated to this function.*
- *All EDSFI closeout team action items will be completed by 12/31/90.*
- *Designs for required modifications will be issued by 3/31/91.*
- *Modifications will be installed during the 1992 refueling outages for each unit.*
- *The NRC inspection report was positive regarding our efforts.*

ENGINEERING DISCREPANCY MANAGEMENT

Design Basis Documentation (DBD) Project - The DBD project is a key to future improvements in management of discrepancies. The DBD Project objectives include:

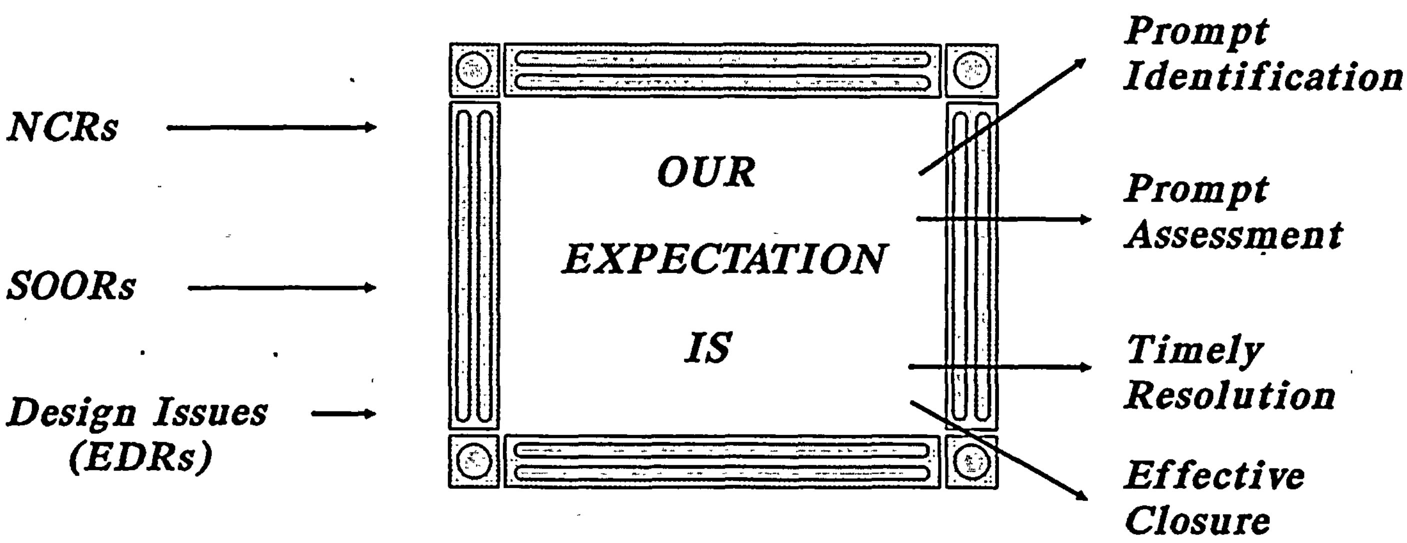
- *Provide a top-level directory of the documents that define the current plant basis and configuration.*
- *Emphasize the design intent (the "why" of the design).*
- *Validate critical design parameters against the design basis.*
- *Index, maintain and control documentation.*

ENGINEERING DISCREPANCY MANAGEMENT

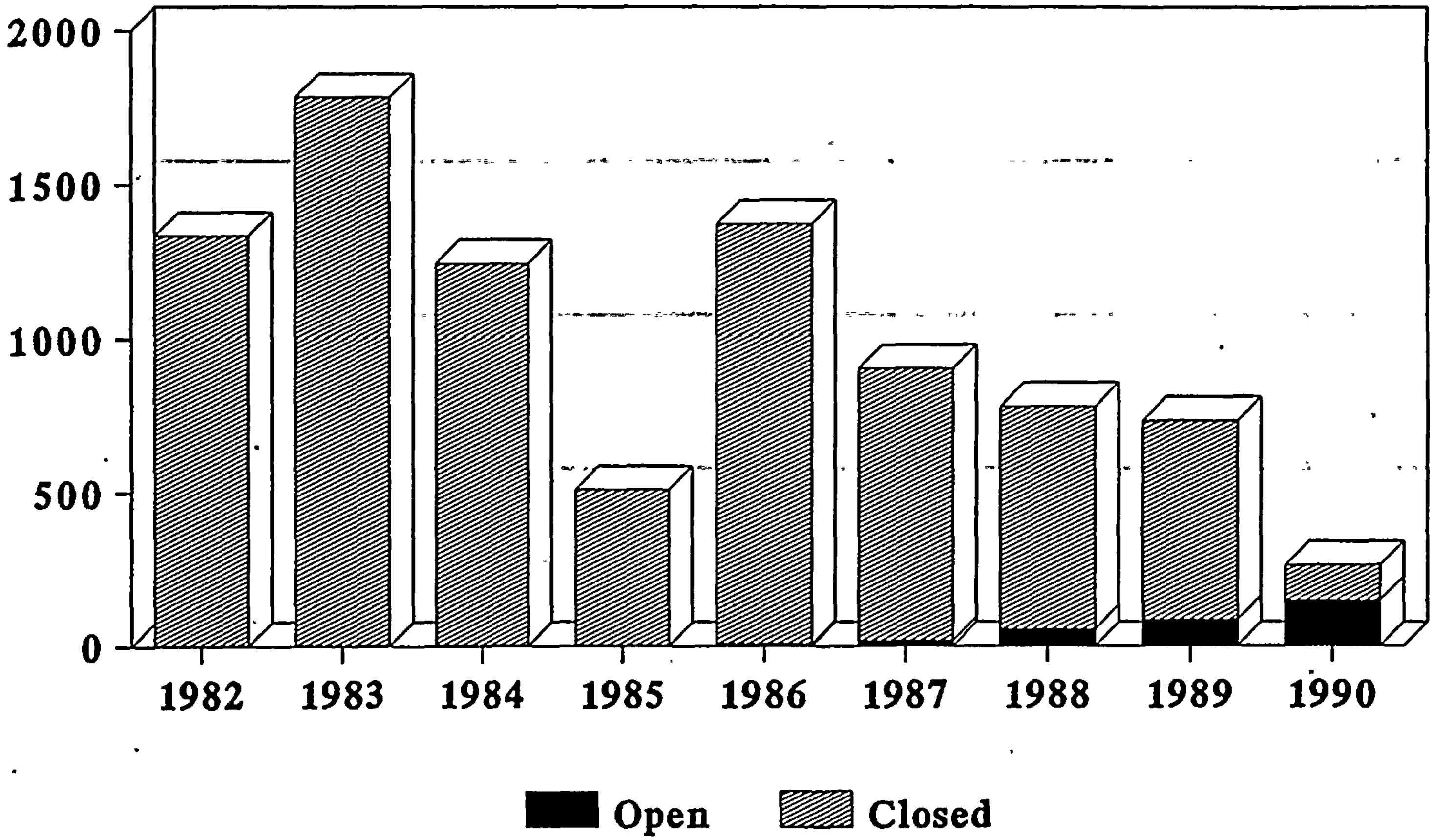
Discrepancy Closure - The entire engineering organization is committed to prompt closure of all safety significant discrepancies.

- *We will retain the open climate on identification of potential discrepancies.*
- *Our focus is on plant safety.*
- *Process improvements will assure proper priority is assigned.*
- *We will strive to close all safety significant discrepancies within one fuel cycle.*
- *Close coordination with the Plant is an integral part of the process and essential for success.*

DISCREPANCY MANAGEMENT



NCR Backlog
Open Vs Closed



As of 10/1/90

DEFICIENCY MANAGEMENT AT SSES

- *Enhanced the program*
- *Added outage scope*
- *Reviewed the backlog and schedule*

PROGRAM ENHANCEMENTS

- *Documentation of basis of operability/ reportability now required on the original*
- *Techniques for documenting operability determinations based on NRC guidance*
  - *Ability to function*
  - *Positive determinations*
  - *Not based on probability*
- *Enhanced up-front handling and review of NCR's*
  - *Communications*
  - *Management involvement*
- *Plant superintendent will approve remaining open NCR's prior to startup*

ADDITIONAL OUTAGE SCOPE

- *Added 31 additional NCR's to Unit 1 outage work scope*
    - *Modifications*
    - *Maintenance*

- *Added 15 additional NCR's to Unit 2 outage work scope*
    - *Modifications*
    - *Maintenance*

- *Anticipate up to 15 more to be added to Unit 2 outage work scope*

DEFICIENCY BACKLOG REVIEW

- *To characterize, understand, prioritize, schedule and close all existing deficiencies (as of 8/13/90)*

- *Three aspects of review*
  - *Re-characterize safety significance*
  - *Re-review operability/reportability*
  - *Affirm status and schedule*

- *Scope*
  - *Phase 1: NCR's, SOOR's, EDR's*
  - *Phase 2: NRC, NSAG, Audit findings*

- *Status*
  - *NCR's, EDR's: Complete*
  - *SOOR's : 11/15/90*
  - *Phase 2: 12/28/90*

NCR BACKLOG REVIEW RESULTS

- *No NCR's required an operability determination change*
- *Assessing 4 NCR's for reportability change*
- *Clarified operability/reportability determinations on 50 NCR's*
- *Re-dispositioned 7 NCR's*

DEFICIENCY BACKLOG REVIEW
SAFETY SIGNIFICANCE CHARACTERIZATION

Category 1 - ***An immediate or continuing condition which impacts continued safe operation of the unit(s) for which there must be compensatory action, and/or shutdown.***

Category 2 - ***A condition which would have impacted continued safe operation of the unit(s) had compensatory actions not been implemented.***

- ***Any deficiency which has existed since prior to 1987 regardless of safety significance.***

- ***All EQ-related items, whether or not merely a paper problem.***

- ***Any item which has been reported via 10CFR50.72/73.***

- ***Condition for which disposition has not been established.***

DEFICIENCY BACKLOG REVIEW

SAFETY SIGNIFICANCE CHARACTERIZATION (CONT'D.)

Category 3 - Condition exists but is analyzed and is acceptable "use-as-is" but it is desirable to restore to conformance with standards.

- ***Any deficiency existing from 1987.***
- ***Condition requires modification which is being developed.***

Category 4 - Condition exists, has no safety significance, and corrective action (modification enhancement, work authorization) is planned and scheduled.

Category 5 - Condition has been restored. Only closeout paperwork remains.

- ***Measuring and testing equipment calibration.***
- ***Procedural changes for enhancement.***
- ***OJT for action to prevent recurrence.***
- ***Scheduled retest of an installed replacement snubber.***

NCR BACKLOG REVIEW UPDATE

10/24/90

277 OPEN NCR'S ON AUGUST 13, 1990

SCHEDULE BREAKDOWN

Closed since August 13th	*81*
Remaining scheduled for U1 5th Refuel Outage	*41*
Scheduled for U2 4th Refuel Outage	*42*
Scheduled for end of year	*58*
Scheduled for 1st quarter 1991	*13*
Schedule under development	*42*

POTENTIAL SIGNIFICANCE BREAKDOWN

CATEGORY 1	*0*
CATEGORY 2	*58*
CATEGORY 3	*46*
CATEGORY 4	*54*
CATEGORY 5	*119*

SOOR BACKLOG REVIEW UPDATE
IN PROGRESS

378 OPEN SOOR'S ON AUGUST 13, 1990

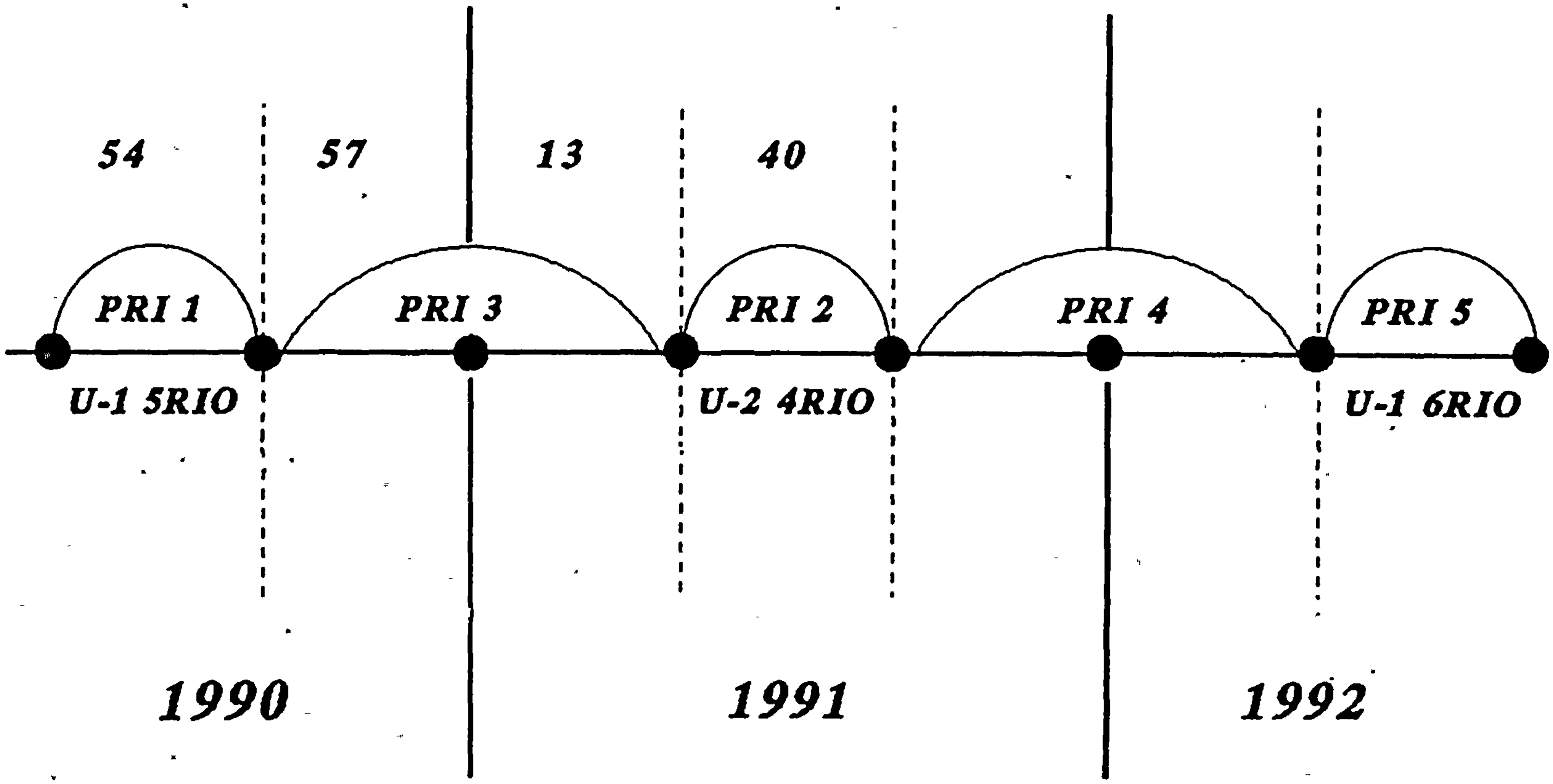
SCHEDULE BREAKDOWN

Closed since August 13th	*118*
Scheduled for U1 5th Refuel Outage	*5*
Scheduled for U2 4th Refuel Outage	*7*
Scheduled for end of year	*43*
Schedule under development	*11*
To be reviewed by 11/15/90	*194*

POTENTIAL SIGNIFICANCE BREAKDOWN

CATEGORY 1	*0*
CATEGORY 2	*5*
CATEGORY 3	*17*
CATEGORY 4	*23*
CATEGORY 5	*139*

NCR SCHEDULE LOGIC



SUMMARY

- ***Backlog is understood***
- ***Backlog reduction is scheduled and in progress***
- ***The system has been changed***

DEPARTMENT DIRECTION

PP&L's Vision: *"To achieve excellence in the operation, maintenance, and support of Susquehanna and, by so doing, be recognized as one of the best nuclear utility organizations in the United States."*

- ***Placing priority on the closing of discrepancies is consistent with our vision.***
- ***We have implemented corrective actions to improve our performance.***
- ***Substantial progress has and will continue to occur.***
- ***PP&L management recognizes that a commitment to effective discrepancy management is for the life of the plant.***