U.S. NUCLEAR REGULATORY COMMISSION

REGION III

REPORT NO. 50-315/96002; 50-316/96002

FACILITY

Donald C. Cook Nuclear Generating Plant

LICENSEE

Indiana Michigan Power Company
Donald C. Cook Nuclear Generating Plant
1 Riverside Plaza
Columbus, OH 43216

DATES

January 17, 1996 through February 26, 1996

INSPECTORS

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APPROVED BY

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3/28/96
Date

AREAS INSPECTED

A routine, unannounced inspection of operations, maintenance, engineering, preparation for refueling, plant support, and review of UFSAR commitments was performed. Safety assessment and quality verification activities were routinely evaluated. Follow-up inspection was also performed for non-routine events.

Executive Summary

OPERATIONS

Human performance issues, which were discussed in previous inspection reports, were again observed during this inspection period as evidenced by:

- Auxiliary equipment operator tours met the minimum requirements defined in the licensee's procedure. However, there appears to be insufficient time being spent to perform the recommended inspections defined in the procedures. (Section 1.1) NRC IDENTIFIED

- Mis-communication among reactor operators contributed to the overload trip of a safety-related valve. Although no damage to the valve was discovered, the event resulted in the licensee being in a TS LCO for a greater period of time. (Section 2.5) SELF-REVEALING

- Licensed operators did not adhere to procedural requirements with regards to auxiliary building integrity controls. (Section 4.2) NRC IDENTIFIED NCV

MAINTENANCE AND SURVEILLANCE

The licensee's root cause investigation and operability determination of the missing residual heat removal system valve gasket were good. (Section 2.4) SELF-REVEALING

The evolution involving the airline replacement of valve 2-QRV-301 was well-planned and coordinated. (Section 2.6) SELF-REVEALING

The licensee continued to have problems with secondary plant material condition which resulted in two plant transients during the inspection period. (Section 1.2) SELF-REVEALING

Two examples of TS surveillance activities were identified where safety-related equipment was not tested under suitably controlled conditions. A violation regarding similar these was issued in a previous inspection report. (Section 2.3) NRC IDENTIFIED

- The surveillance test for the turbine-driven auxiliary pumps required the operability of steam traps to be checked prior to starting the pumps. The procedure did not require that an operability evaluation of the as-found condition be performed if a malfunctioning steam trap was discovered and bypassed.

- The licensee occasionally lubricated and exercised fuel rack linkages prior to performing TS monthly emergency diesel generator surveillances.

Failure of electrical maintenance personnel to comply with procedural requirements resulted in the overload trip of a safety-related valve.

Although no damage to the valve was discovered, the event resulted in the licensee being in a TS LCO for a greater period of time. Concerns regarding adherence to plant procedures have been discussed in previous inspection reports. (Section 2.5) SELF-REVEALING

ENGINEERING

The licensee installed a tarp over a portion of the Unit 2 refueling cavity which was not in accordance with a previously approved safety evaluation. Concerns regarding weaknesses in licensee safety evaluations have been discussed in previous inspection reports. (Section 3.1) NRC IDENTIFIED NOV

The licensee did not properly install the new fuel vault criticality monitor system to meet 10CFR70.24 requirements. (Section 4.1) NRC IDENTIFIED NOV

Concerns were identified regarding foreign material exclusion control of new and spent fuel. Similar concerns have been discussed in previous inspection reports. (Section 4.3) NRC IDENTIFIED

PLANT SUPPORT

There was excellent dose performance during the fall 1995 refueling outage and good planning for the spring 1996 refueling outage. Collective dose during 1995 was low. (Section 5.0)

SAFETY ASSESSMENT AND QUALITY VERIFICATION

Licensee personnel exited a Technical Specification (TS) action statement for intermediate deck ice condenser doors while plant conditions did not meet the limiting condition for operation (LCO). A similar concern regarding non-conservative interpretation of TSs was discussed in a previous inspection report. (Section 2.2) LICENSEE IDENTIFIED

The licensee took appropriate action to address a NRC concern regarding the resetting of time in the LCO action statement on a daily basis for performance of an ice condenser surveillance. (Section 2.2) NRC/LICENSEE IDENTIFIED

Some minor discrepancies in the UFSAR were identified. (Section 6.0) NRC IDENTIFIED

Summary of Open Items
Violations: identified in Section 3.1 and 4.1
Unresolved Items: identified in Section 4.1
Inspector Follow-up Items: identified in Section 6.0
Non-cited Violations: identified in Section 2.5 and 4.2

INSPECTION DETAILS

1.0 OPERATIONS

NRC Inspection Procedure 71707 was used in ongoing inspection of plant operations. The NRC continue's to have concerns with the quality of auxiliary operator rounds, human performance, and procedural adherence.

1.1 Auxiliary Equipment Operator (AEO) Tours - Both Units

While the AEO tours met all licensee and regulatory requirements, the tours were determined by the NRC to meet the minimum requirements of the tour procedure. However, there appears to be insufficient time being spent to perform the recommended inspections defined in the procedures.

Previous information concerning AEO tour completeness and quality were documented in NRC Inspection Reports 50-315/316-93019, 93024, 94002, and 94014. As documented in report 315/316-94002 a licensee internal audit had identified that, based on the time the AEOs spent performing tours in certain rooms, a complete and thorough tour was generally not being performed. As part of the corrective action to a previous concern regarding the quality of AEO tours, the licensee issued procedural requirements governing tour performance. This guidance required that AEOs carry log sheets to document certain readings and also that shift management periodically accompany the AEOs during plant tours.

The NRC accompanied AEOs on tours and determined that the required rooms were being entered and the required information was being recorded. However, the operators were not spending a significant amount of time observing the recommended activities provided in the tour procedure. These recommended activities included, checking valve alignments, V-belt condition, shear pins not broken, basin drains free of debris, differential pressure not excessive, etc.

In addition, the NRC reviewed several licensee tour verification sheets. These reviews also determined the tours were generally of such a duration that the AEOs were performing only the minimum required by procedure. These sheets were used by shift management to verify, on a sample basis, that the operators were entering the required rooms and sufficient time was being spent in the toured areas.

The licensee agreed to evaluate the NRC observations to ensure that management expectations were being met by the AEOs.

1.2 Secondary Side Transients - Unit 2

As discussed in NRC Inspection Report 50-315/316-95012 the licensee had experienced a number of secondary side transients due to the poor material condition of some secondary side components. The material condition of the secondary side has not improved and continued to result in secondary side transients.

During this assessment period the licensee experienced two secondary side transients. Both of the transients involved the Unit 2 heater drain pumps.

- During a January 27, 1996 scheduled level test on the 5B feedwater heater, the normal level controller appeared to stick causing a loss of heater level and the tripping of the South and Middle heater drain pumps. The Middle condensate booster pump and the East turbine auxiliary cooling water (TACW) pump automatically started and the operators manually started the South hotwell pump to re-establish heater levels. The auto start of the standby pumps combined with operator action limited the scope of the resulting main feedwater pump suction pressure transient.

- On February 2, 1996, while the Unit 2 South heater drain pump (HDP) was being restored to service following maintenance, controller problems resulted in the minimum flow valve not fully closing automatically after the pump was started. The extra flow caused heater drain tank levels to drop to the low setpoint and caused the South and the Middle HDPs to trip. The North hotwell pump, the Middle condensate booster pump, and the West TACW pump automatically started as designed. The auto start of the standby pumps combined with operator action limited the scope of the resulting main feedwater pump suction pressure transient.

1.3 Operator Licensing Initial Examination

During the week of January 8, 1996, one initial operator licensing examination was administered to an SRO instant candidate. The candidate demonstrated overall weak performance, particularly on the written examination. The licensee, in general, demonstrated difficulty in developing examination materials for the pilot program, particularly in written test question development, job performance measure verification, and operating test validation.

2.0 MAINTENANCE AND SURVEILLANCE

NRC Inspection Procedures 62703, 61726, and 92902 were used to perform an inspection of maintenance and testing activities. Although the NRC had concerns with the licensee's performance of ice condenser surveillances, the licensee's concurrent identification of these concerns and the resulting actions were viewed as very positive. In addition, the licensee's root cause investigation and operability determination of the residual heat removal system valve gasket failure was also good. However, the NRC identified examples where safety-related components were not being tested under suitably controlled conditions. In addition, the NRC continued to have concerns regarding non-conservative interpretations of TS requirements, procedural adherence, and secondary plant material condition.

2.1 Maintenance and Surveillance Testing Activities

The NRC observed routine preventive and corrective maintenance and surveillance activities to ascertain that these were conducted in accordance with approved procedures, regulatory guides, industry codes or standards, and in conformance with Technical Specifications (TS). The specific items observed/reviewed are listed below:

Maintenance Activity	Description
C0034641	Containment polar crane - clean and inspect drum gears
C0033468	Clean and paint polar crane trolley rails
C0033923	Containment polar crane - inspect main hoist mechanical load break
R0054374	1-CD-EDG, Lubricate and exercise the fuel racks.
R0040929	1-AB-EDG, Lubricate and exercise the fuel racks.
R0053727	1-AB-EDG, Lubricate and exercise the fuel racks.
C0033810	Repair 1-IRV-311

Surveillance Activity	Description
EHP.4030.STP.211	Ice Condenser Surveillance
THP.4030.STP.245	Inspection of Ice Condenser Intermediate Deck Doors
EHP.4030.STP.250	Inspection of Ice Condenser Flow Passages
OHP 4030STP.017T	Turbine Driven Auxiliary Feedwater System Test
IHP 5021.EMP.004	Limitorque Limit And Torque Switch Setting
OHP 4030.STP.041	Refueling Integrity Verification

2.2 Ice Condenser Technical Specifications (TS) Action Statement Error - Unit 2

On February 22, 1996, with Unit 2 at 100 percent power (Mode 1), the licensee began performance of the TS 18-month surveillances (4.6.5.3.1(b)) on the Unit 2 ice condenser doors in preparation for the upcoming refueling outage. This required entry into the action statement for TS 3.6.5.3 due to one or more intermediate deck doors (IDDs) being inoperable. The action statement allowed this condition to exist for up to 48 hours with no action, or 14 days provided that ice bed temperatures were monitored every 4 hours and maintained under 27°F. The licensee's intent was to exit the action statement at the end of each day as work was only being performed on the day shift. Therefore, the time out of service and in the LCO would have been reset each day.

At the end of day shift (4:30 pm) on February 22, 1996, surveillance activities were stopped and the lead engineer reported to the control room that all IDDs were returned to service except door A in bay 9. This door failed the required load test. The engineer also discussed a memo from corporate engineering which stated that the ice condenser was capable of performing the required safety function with up to 19 IDDs inoperable. Based on the engineering memo, the unit supervisor declared the ice condenser operable and exited TS 3.6.5.3. The shift technical advisor (STA) and

assistant shift supervisor reviewed the condition report documenting the inoperable IDD and concurred that entry into an action statement for one inoperable IDD was not required.

Since one IDD was inoperable, TS 3.6.5.3 should not have been exited. While reviewing the condition report generated for the inoperable IDD, the oncoming night shift STA identified that the unit should still have been in TS 3.6.5.3, and the action statement was re-entered at 7:30 pm. No TS action statement time limitations were exceeded due to this error.

The NRC had the following concerns with this evolution:

- Neither the operators nor the engineers verified that the licensing bases for the ice condenser were met. Both organizations allowed guidance contained in an engineering memo to supersede TS requirements. A similar event concerning diesel generator TS requirements was discussed in NRC Inspection Report 50-315/316-95013.

- The intended practice of resetting the time in the action statement by entering and exiting the LCO on a daily basis was considered a non-conservative approach as cumulative out-of-service time would not have been considered.

- The licensee did not utilize the plant's probablistic risk assessment (PRA) to provide insight into the impact of performing ice condenser surveillances while at power. The NRC review determined that these activities did not present an increase in risk because the ability of the ice condenser to perform the required safety function was not impacted

The NRC viewed the licensee's response to this occurrence as positive. The oncoming shift identified the erroneous TS interpretation, and further performance of the surveillances was stopped until the appropriate course of action was determined. Also, licensee management recognized the significance of this event in light of a similar event involving a missed diesel generator surveillance. Both instances involved using information other than that contained in the licensing basis to determine the applicability of TS requirements.

Licensee senior management had similar concerns regarding the management of LCO time and had been reviewing this matter prior to this event. Prior to resuming performance of the surveillances, the licensee established a limit of 200 hours of cumulative time in the action statement was set to ensure compliance with the intent of TS.

Additional licensee actions included; issuing a memo to all operations shifts describing the details of the incident, and the Operations Superintendent expressing his expectation to all shifts emphasizing that licensed operators must be responsible for ensuring compliance with licensing requirements.

2.3 Pre-Conditioning of Equipment Prior to Surveillance Tests - Both Units

As documented in NRC Inspection Report 50-315/316-95013(DRP), the NRC previously identified examples of TS surveillance activities where safety-related equipment was not tested under suitably controlled conditions and in accordance with design and licensing bases. During the latest inspection period, the NRC identified two additional examples of pre-conditioning.

- During routine review of surveillance procedure OHP 4030STP.017T, "Turbine Driven Auxiliary Feedwater System Test," the NRC noted that the licensee verified the operability of steam traps associated with the steam supply to the pump turbine prior to starting the pump. The licensee performed the test by placing a listening device on each trap discharge pipe and checking for proper operation. If no flow was heard, the trap was determined to be malfunctioning, and the procedure required the trap bypass valve be opened to provide a continuous flow path. The procedure also required that an action request be issued to repair the trap.

  The NRC concluded that the licensee's actions were prudent to verify the steam traps were operable to prevent condensate from damaging the turbine. However, the NRC were concerned that the procedure did not require that a condition report be initiated to address the operability of the pump in the "as found" condition if a malfunctioning trap was identified. The NRC noted that the traps that were tested were required to be operable or bypassed to support TDAFW pump operability.

  The NRC did not have any immediate operability concerns, due to the reliability of the traps and the few failures that have been experienced. The NRC reviewed surveillance records and discovered only one recent example of a malfunctioning trap. Action Request (AR) A0085828, dated December 8, 1994, documented that no flow was detected at the outlet of drain 26 for the Unit 2 TDAFW pump and the bypass valve was throttled open as required by procedure. However, no operability evaluation was performed.

- The NRC identified that an emergency diesel generator (EDG) preventive maintenance (PM) activity, to lubricate and manually exercised the linkage of each fuel injection pump to ensure there was no binding or sticking, was performed every 30 days. The NRC noted that licensee management was unaware of the purpose and the frequency of the PM on the linkage. The NRC was concerned that controls had not been established to prevent performance of the PM prior to running the EDGs during the monthly TS surveillance test. The pre-conditioning of the fuel injection pump linkages prior to surveillance testing could mask a problem that would prevent the EDG from performing as designed.

The NRC reviewed licensee records and identified three examples where the licensee exercised the fuel linkages within 24 hours of running the EDG:

1-CD-EDG on January 23, 1996 (R0054374)
1-AB-EDG on February 7/8, 1995 (R0040929)
1-AB-EDG on January 9/10, 1996 (R0053727)

The NRC reviewed the job orders and verified that no abnormalities were documented that would have prevented the EDGs from starting as designed. However, the NRC were concerned that the licensee had not established procedures to evaluate the as-found operability of the EDGs if non-conforming or degraded conditions were identified during the PM activities.

These examples of failure to test equipment under suitably controlled conditions represent a violation of NRC requirements. However, a notice of violation will not be issued since these are further examples of a previously identified violation (documented in NRC Inspection Report 50-315/316-95013) for which the licensee has not had the opportunity to fully implement corrective actions.

2.4 Residual Heat Removal (RHR) Valve Gasket Leak - Unit 1

On January 31, 1996, while disassembling the RHR heat exchanger bypass control valve, 1-IRV-311, to repair a flange leak, the licensee discovered that approximately 50 percent of the flexitallic gasket material was missing.

As discussed in NRC Inspection Report 50-315/316-95010, the licensee replaced the gasket in August 1995 after discovering pieces of gasket material from a previous gasket failure during the last refueling outage. At that time, the licensee performed full flow flushes of the emergency core cooling system (ECCS) pumps' discharge piping to remove any material remaining in those lines. In addition, due to a concern that material in the suction piping could potentially damage the safety injection and centrifugal charging pumps, the licensee installed a cleanout connection and removed gasket material found in the suction header. The licensee determined that the root cause of the first failure was the installation of an undersized gasket during maintenance on the valve in 1994.

In response to the latest identified failure, the licensee performed a prompt operability determination, as documented in Condition Report No. 0127, to address the potential damage to the ECCS pumps. The licensee concluded that the gasket probably failed during the flushes performed during the previous refueling outage. The basis for the conclusion was that, unlike the previous cycle, no indication of fuel damage has been detected since the beginning of the present cycle and that this portion of the line did not normally receive flow. The licensee had determined that the first gasket failure caused the fuel failures during the previous cycle.

The licensee's backup operability determination, dated February 2, 1996, concluded that, in the unlikely event gasket material migrated to the ECCS suction header, the pumps would not be damaged. This was based on a review of

the piping configuration and operating experience provided by the pump vendor of similar pumps in fossil plants.

The licensee determined that the root cause of the latest failure was the misalignment of the gasket and the valve discharge flange during installation. The valve was maintained in place between two pipe flanges and centering of the gasket was critical due to the tight tolerances between the outside diameters of the valve discharge flange and the gasket filler element. The problem was compounded because the pipe flanges were not concentric with each other. As immediate corrective action, the licensee replaced the flexitallic gaskets with a compressed fiber gasket, custom sized to ensure proper alignment. On a long-term basis, the licensee will determine other alternative gasket designs. The licensee also intended to inspect the 2-IRV-311 valve gasket prior to the upcoming Unit 2 refueling outage. The NRC concluded that the licensee's root cause investigation and operability determination were good.

2.5 Valve Overload Trip - Unit 2

On February 21, 1996, while restoring the out-of service clearance on the charging pump suction cross-tie valve (2-IMO-361), the valve traveled in the direction opposite than what was expected and the feeder breaker tripped on overload. Following motor operator replacement, electrical maintenance had signed off on the out-of-service clearance but verbally requested operations shift management to contact them prior to operating the valve. The purpose for the request was to verify the correct rotation of the motor.

However, the unit supervisor (US) did not effectively communicate the request to the reactor operator (RO) during a general announcement to the operating crew. The RO was busy and did not respond positively to the command but later took the valve control switch to the closed position, as required by the clearance restoration, without notifying the electricians. Following the event, the licensee performed an internal inspection of the valve and no damage was noted, however, the event resulted in the licensee being in a TS LCO for a greater period of time.

During review of the Condition Report (CR), the NRC noted that paragraph 7.4.1 of procedure 12 IHP5021.EMP.004, "Limitorque Limit And Torque Switch Setting," required that a deadman switch or equivalent be installed prior to releasing the clearance permit. The deadman would have prevented operation of the valve from the control room. The failure to comply with the procedure constituted a violation of minor significance and is being treated as a Non-Cited Violation (50-316/96002-04), consistent with Section IV of the NRC Enforcement Policy. As immediate corrective action, the licensee enhanced the applicable procedures and briefed personnel on the importance of adhering to procedures.

2.6 Replacement of Airline to Valve QRV-301 - Unit 2

The NRC observed the evolution involving the airline replacement of the letdown backpressure regulating valve (2-QRV-301), and concluded that the evolution was well-planned and implemented. The evolution required that the valve be isolated, while maintaining letdown pressure using a manually

operated bypass valve. The evolution required coordination between the reactor operator and the auxiliary operator who was operating the manual valve locally. The NRC observed that the prebrief and communications among the involved parties was good.

3.0 ENGINEERING

NRC Inspection Procedures 37550 and 37551 were used to perform an onsite inspection of the engineering functions. The NRC identified a failure to perform a 50.59 evaluation and weaknesses with other 50.59 evaluations. In addition, as noted in paragraph 4.0, there was a weakness of reactor engineering not taking "ownership" in the receipt and storage of new fuel, and concerns with foreign material exclusion.

3.1 Work In Containment While At Power - Unit 2

In an effort to reduce the refueling outage scope, the licensee was performing more maintenance and modifications while on line. The NRC determined that while the work was being performed in a safe manner it was not always fully or properly evaluated prior to the work being started. The NRC identified concerns with potential debris and with a tarp installed inside containment without a safety evaluation.

The NRC observed that in an effort to keep debris out of the containment refuel pool a debris/cargo net had been installed over the pool. The NRC verified that a safety evaluation had been performed and that the nets were installed in accordance with the evaluations. However the NRC identified weaknesses with the evaluations.

- A safety evaluation had been performed for the installation of a jib crane in Unit 2. In order to allow the installation of the nets, the evaluation incorporated by reference an evaluation for previously performed Unit 1 work. The licensee spent considerable effort to verify that the Unit 1 evaluation could be utilized for the Unit 2 work. This method met the regulatory requirements but was cumbersome. This increased the opportunities for errors.

- In response to NRC questions, the licensee located design documents showing that the ice condenser top deck doors would not be affected by movement of the polar crane and that the crane met seismic requirements even when moved from its parked position. Discussion of these issues was not included in the safety evaluations.

While inside the containment the NRC noted that workers properly secured all bags of material as required by procedures and the safety evaluations. However, during movement, about a dozen bags became unsecured. While the bags were attended at all times, during a Loss of Coolant Accident (LOCA) the workers could have left the bags in place during the containment evacuation. In response to this concern the licensee instructed the radiation protection technicians to limit large numbers of unsecured bags during containment tours.

During a subsequent tour of containment, the NRC noted a 8 ft by 8 ft yellow tarp secured over the refuel pool which was not in accordance with the approved safety evaluation. The licensee responded by removing the tarp and performing an analysis of the as found configuration. The licensee determined that the containment lower drains and the recirculation sumps were still operable. However, the licensee's analysis initially failed to consider the effects of post LOCA pH on the tarp and its supporting rope until questioned by the NRC. The failure to perform a safety evaluation as required by 10 CFR 50.59 prior to installing the tarp is a violation 316/96002-02(DRP).
In response to the identification of the tarp the plant manager issued a stop work order for all work in the Unit 2 containment. Prior to restarting the work, all workers were retrained on the potential consequences of loose debris in the containment. In addition to the above corrective action, the containment system engineer was tasked with the responsibility of maintaining cognizance of all work being performed inside containment while the unit was on line.

4.0 PREPARATION FOR REFUELING - Unit 2

NRC Inspection Procedure 60705 was used to perform an inspection of the licensee's preparation for the planned Unit 2 refueling outage. This inspection primarily focused on the control of and movement of new fuel during the receipt inspection. The NRC determined that while the receipt, storage and handling of the new fuel was safe, there were failures to meet regulatory requirements and a lack of ownership over new fuel.

4.1 Operability of The New Fuel Vault Criticality Monitor - Unit 2

10 CFR Part 70.24 required, in part, that the licensee maintain a monitoring system capable of detecting a criticality in the new fuel storage vault (NFV). The monitoring capability was required even though licensee calculations show that a criticality event in the new fuel vault was not credible. Following a routine tour of the new fuel storage vault, the NRC questioned the ability of the installed monitoring system to comply with 70.24. Subsequent licensee calculations and monitor setpoint determinations showed that the installed monitoring system would not meet the requirements of either 70.24(a)(1) or 70.24(a)(2).

During a routine inspection of new fuel receipt and of the new fuel vault, the NRC questioned the ability of the one installed radiation monitor to meet the requirements of 10 CFR Part 70.24. The monitor was located outside the vault and was separated from the closest fuel assembly in the new fuel vault by a distance of approximately 30 feet which included one 18" thick reinforced concrete wall. Subpart (a)(2) of 70.24 required the monitor be capable of detecting a 300 R/hr field that was one foot from the fuel. In addition, (a)(2) required that the monitoring devices have a set point of not less than 5 mr/Hr nor more than 20 mR/hr. Subpart (a)(1) of 70.24 required, in part, two radiation monitors for the new fuel.

The NRC was concerned that the distance to the monitor combined with the thick wall would defeat the purpose of the one installed monitor. Interviews with licensee personnel and the review of licensee documents determined:

- The licensee was not knowledgeable of the need to meet the requirements of 70.24.

- The licensee had information regarding an exemption to 70.24(a)(2) that had been granted in the mid 1970s, but the information indicated the exemption had lapsed.

- The monitor had not been installed in accordance with an approved design change package but was instead a temporary monitor that had been in place since December 2, 1993. Prior to that time, another temporary monitor had been installed in a different location.

- The calculation which supported installation of the monitor in 1993, contained assumptions as to locations and distances which were not met. Specifically the monitor was assumed to be located outside of the new fuel vault some 9 feet above the fuel. Instead the monitor was located outside the new fuel vault some 30 feet horizontally from the fuel. The increased distance caused the radiation field to be seen by the monitor to drop below that required by 70.24(a)(2). The NRC determined that no licensee procedures governed the placement of the criticality monitor.

- In response to an NRC request, the licensee checked the actual setpoint of the monitor. The monitor was determined to be set to alarm at 1,000 mr/hr. 70.24(a)(2) required the monitor to be set between 5 and 20 mr/hr. The NRC determined that no plant procedures governed the setpoint of the criticality monitor.

- A review of the licensee's USAR identified that the NFV criticality monitor was not addressed in any section. Area radiation monitors/criticality monitoring devices were a part of the licensee's design and licensing basis that should be in the USAR.

- The licensee had initially assumed they were required to meet 70.24(a)(2) for Unit 2. Subsequently, the licensee and the NRC questioned whether (a)(2) or (a)(1) was the appropriate requirement. This issue will be a part of the unresolved item discussed below.

The licensee's failure to have a radiation monitor that met the requirements of 10 CFR Part 70.24 is considered a violation (50-316/96002-01(DRP)).

The NRC also had questions concerning the licensee's emergency procedures for a criticality event, the need for drills, and the need for either constant monitoring or a detector which would send signals to a remotely monitored location. NRC resolution of these questions and whether the licensee must comply with 70.24(a)(1) or (a)(2) is an Unresolved Item (50-316/96002-03(DRP)).

The licensee's initial response to this issue was slow and not focused. Initially there were no attempts to expand the questions concerning detector operability beyond those raised by the NRC. This occurred even though new fuel was continuing to be stored and loaded into the new fuel vault.

Following the initial inoperability determination the licensee made a one hour phone call to the NRC. To restore the detector to operable the licensee repositioned one new fuel assembly so that the detector would have a direct line of sight. At that point the licensee believed the detector had been restored to operable. No further questions were asked by the licensee concerning detector operability until the NRC questioned the setpoint of the detector. At that point the licensee determined that the detector was inoperable due to the wrong setpoint and another 1 hour report was made to the NRC.

4.2 Poorly Implemented Auxiliary Building Integrity Controls - Unit 2

On February 27, 1996, the NRC identified that restoration of auxiliary building integrity (ABI) had not been verified prior to performing fuel movement in the spent fuel pool (SFP).

Following the off load of the new fuel transportation canisters in the auxiliary building, the senior reactor operator-core alterations (SRO-CA) attempted to close the crane-bay door but, the door remained approximately 1-2 feet open. The SRO-CA left the area and made a mental note that the door would be closed prior to moving fuel in the SFP.

During a routine plant tour, the NRC noted the unattended, partially open door while new fuel was being unloaded from transportation canisters and inspection activities were about to begin. The NRC notified the licensee and the bay door was immediately closed.

Operations procedure 01-OHP-4030.STP.041, "Refueling Integrity," Revision 4, provided the controls necessary to ensure that ABI was maintained while fuel movement was in progress. A note in the procedure stated that "when a piece of equipment is out of position, not to include normal passage, this fact must be logged on Data Sheet No. 4, Loss of Refueling Integrity Log." Step 4.9 further states, "before refueling or spent fuel pit operations commence, verify restoration of all items on Data Sheet No. 4."

Due to the NRC's questions, the appropriate Data Sheet entries were made for the door being out of position. The failure to document the partially open door on data sheet 4 and failure to verify restoration of all items on the data sheet is considered a violation of procedure STP.041. However, since no fuel movement occurred with the door partially open, this failure constitutes a violation of minor significance and is being treated as a Non-Cited violation, consistent with Section IV of the NRC Enforcement Policy 50-316/96002-05.

4.3 Foreign Material Exclusion (FME) For New Fuel and the Spent Fuel Pool - Unit 2

The licensee and the NRC identified examples where foreign material was allowed to be in contact with new and spent fuel. Problems with FME were identified in NRC inspection report 50-315/316-95010(DRP) and was the subject of inspection follow-up item (IFI) (50-315/316-94024-02(DRP)). As was noted

in the 315/316-95010 report, FME controls were improving but more attention in this area was needed.

During the receipt and inspection of the new fuel, the licensee identified two examples where foreign material inadvertently reached some assemblies. Those examples were:

- On February 9, 1996, oil leaked onto two assemblies when the seal on a scale failed. The assemblies were not damaged but did have to be returned to the factory for cleaning.

- On February 26, 1996, a safety clip on a chain hook being used to lift the fuel into the vertical position broke. Several small pieces of metal landed on top of the fuel and on the floor around the fuel. Fuel movement was immediately halted. Licensee personnel performed a detailed evaluation of the clip and determined that the head of one small screw was missing. A maintenance mechanic found the screw head on the upper most grid support inside one of the assemblies. The screw head was removed and reverification made that no other parts were missing.

In addition, during routine observations of the spent fuel pool on January 27, 1996, the NRC identified some material on top of old fuel and some material on the bottom of the pool. The material on the bottom of the pool consisted of resin, a washer, and a piece of stainless steel. The resin was left over from an old occurrence where resin was inadvertently backflushed from the spent fuel pool clean up system. The pieces of metal and the material on top of the used assembly were relatively recent. A regular inspection of the spent fuel pool performed on January 18, 1996 by reactor engineering revealed that no material was in the pool. The licensee initiated action requests to have the material removed from the top of the fuel and the bottom of the pool and initiated a condition report (96-0293). The licensee intended to vacuum up the resin; however, it could not be removed from the unaccessible areas of the storage racks and additional material would gradually migrate back out.

5.0 PLANT SUPPORT

NRC Inspection Procedure 84750 was used to perform an inspection of plant support activities. As noted in paragraph 4.0 above, the licensee failed to ensure that a radiation monitor was installed and operated in accordance with 10 CFR 70.24. In addition, NRC review of licensee performance regarding dose for the upcoming unit 2 refueling outage and for the previous unit 1 refueling outage showed:

- The 1995 collective dose was low (202 person-rem).

- There was excellent dose performance during the fall 1995 refueling outage and good planning for the spring 1996 refueling outage.

6.0 Review of UFSAR Commitments

A recent discovery of a licensee operating a facility in a manner contrary to the Updated Final Safety Analysis Report (UFSAR) description highlighted the need for a special focused review that compared plant practices, procedures and/or parameters to the UFSAR description. While performing the inspections discussed in this report, the NRC reviewed the applicable portions of the UFSAR that related to the areas inspected. The following inconsistencies were noted between the wording of the UFSAR and the plant practices, procedures, and/or parameters observed by the NRC.

- The placement of a tarp inside the Unit 2 containment was performed without a review of the UFSAR commitments (paragraph 3.1).

- Section 6.10 of the UFSAR stated that "alarms and redundant level indicators are provided in the containment recirculation sump...." In addition, section 7.5, stated that "the lower range channels indicate the water level in the containment sump." The containment sump water level was also stated in table 7.8-2 as reading out in the control room. The NRC determined the licensee removed the recirculation sump level indicators and moved them to the adjacent sump which is connected. The licensee failed to properly update all pertinent sections of the UFSAR at the time of the modification. (50-315/316-96002-06)

7.0 Meetings and Other Activities

a. Management Meetings

On January 11, 1996, there was management meeting between NRC and the licensee to discuss the personnel performance issues during the Unit 1 refueling outage and during subsequent Unit 1 and Unit 2 reactor operations. In addition, the poor material condition of certain secondary side components and the causes were also discussed. A copy of the material used by the licensee to discuss these matters is attached.

b. Exit Meeting

The NRC contacted various licensee operations, maintenance, engineering, and plant support personnel throughout the inspection period. Senior personnel are listed below.

At the conclusion of the inspection on March 7, 1996, the NRC met with licensee representatives (denoted by *) and summarized the scope and findings of the inspection activities. The licensee did not identify any of the documents or processes reviewed by the NRC as proprietary.

*A. Blind, Site Vice President
*J. Sampson, Plant Manager
*K. Baker, Assistant Plant Manager
D. Noble, Radiation Protection Superintendent
*T. Postlewait, Site Engineering Support Manager

*J. Wiebe, Superintendent, Plant Performance Assurance
*M. Barfelz, Superintendent, Nuclear Safety & Analysis
*J. Allard, Maintenance Superintendent
*L. Gibson, Manager of Business Performance
*S. Colvis, Licensing Engineer
*D. Morey, Chemistry Superintendent
*P. Schoepf, Plant Engineering Superintendent
*T. Beilman, Scheduling Superintendent
*S. Hover, Licensing Engineer
*B. Burgess, Information Communications Services
*R. Leonard, Plant Engineering - System Engineer
*R. Smith, Plant Engineering - System Engineer
*L. Smart, Licensing Engineer
*S. Brewer, Manager Regulatory Affairs
*A. Verteramo, Reactor Engineering Supervisor, Plant Engineering
*P. Russell, Fire Protection Supervisor
*M. Ackerman, Licensing Engineer
*J. Cassidy, Radiation Protection
*L. VanGinhoven, Material Management Superintendent
*M. Depuydt, Licensing Coordinator
*R. West, Licensing Coordinator
*R. Gillespie, Operations Superintendent

Attachment: American Electric Power Agenda

AMERICAN ELECTRIC POWER

January 11, 1996

Agenda

Opening Remarks

Al Blind - Plant Manager/Site Vice President

Introduction

John Sampson - Assistant Plant Manager

Operations

Bill Nichols - Acting Operations Superintendent

Scheduling

Terry Beilman - Scheduling Superintendent

Maintenance

John Allard - Maintenance Superintendent

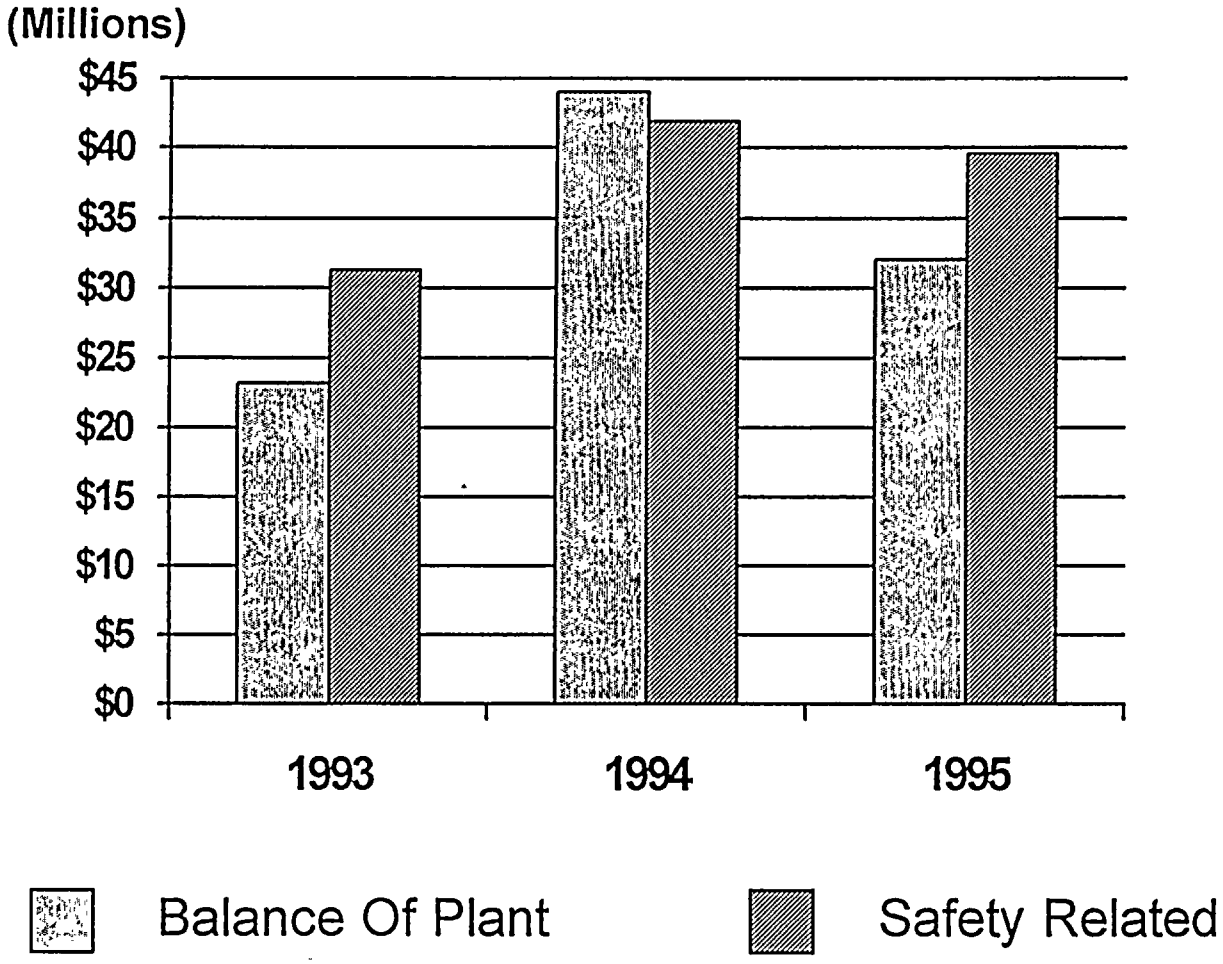
Plant Engineering

Paul Schoepf - Plant Engineering Superintendent

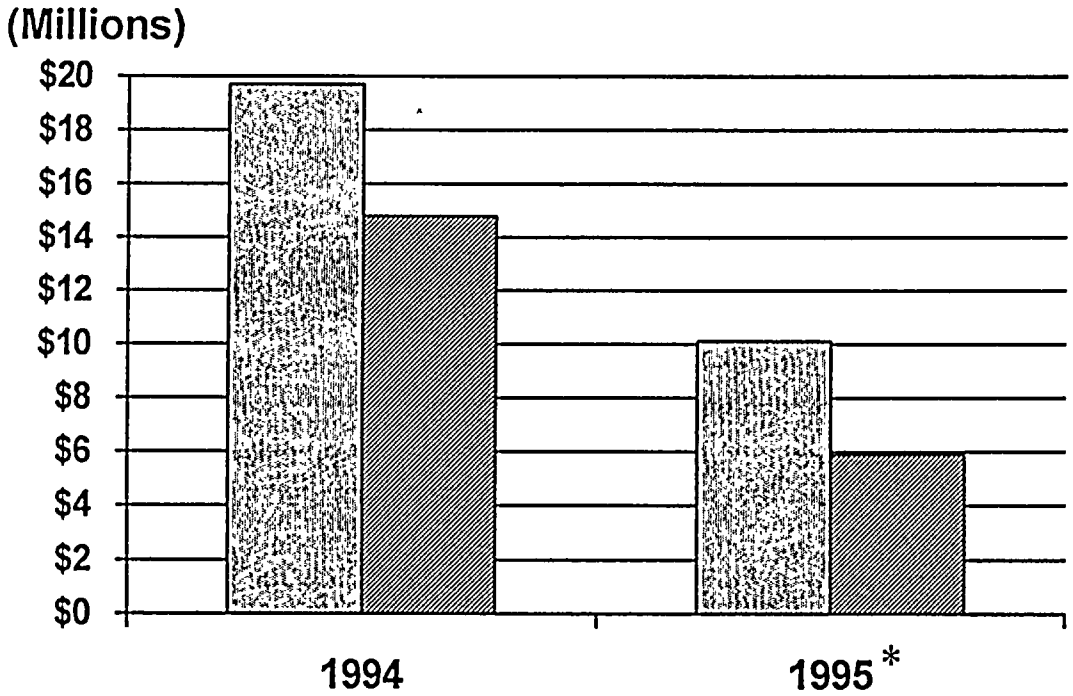
Closing Remarks

Gene Fitzpatrick - Senior Vice President

Maintenance Costs



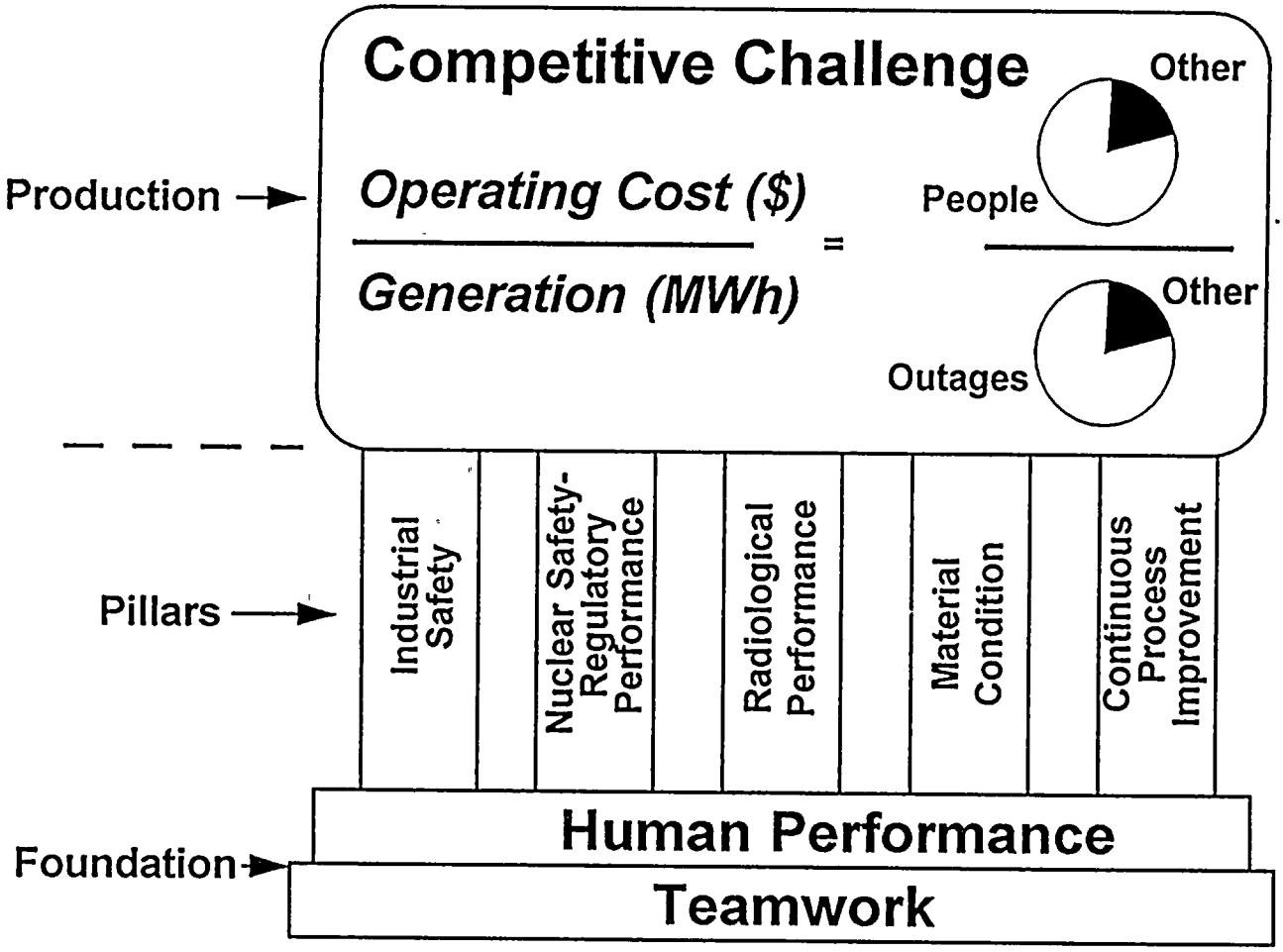
Capital Costs



Balance Of Plant Safety Related

*1995 Capital represents only 11 months (data lag)

Business Model



FUNDAMENTAL ISSUES

Human Performance

Operational Focus

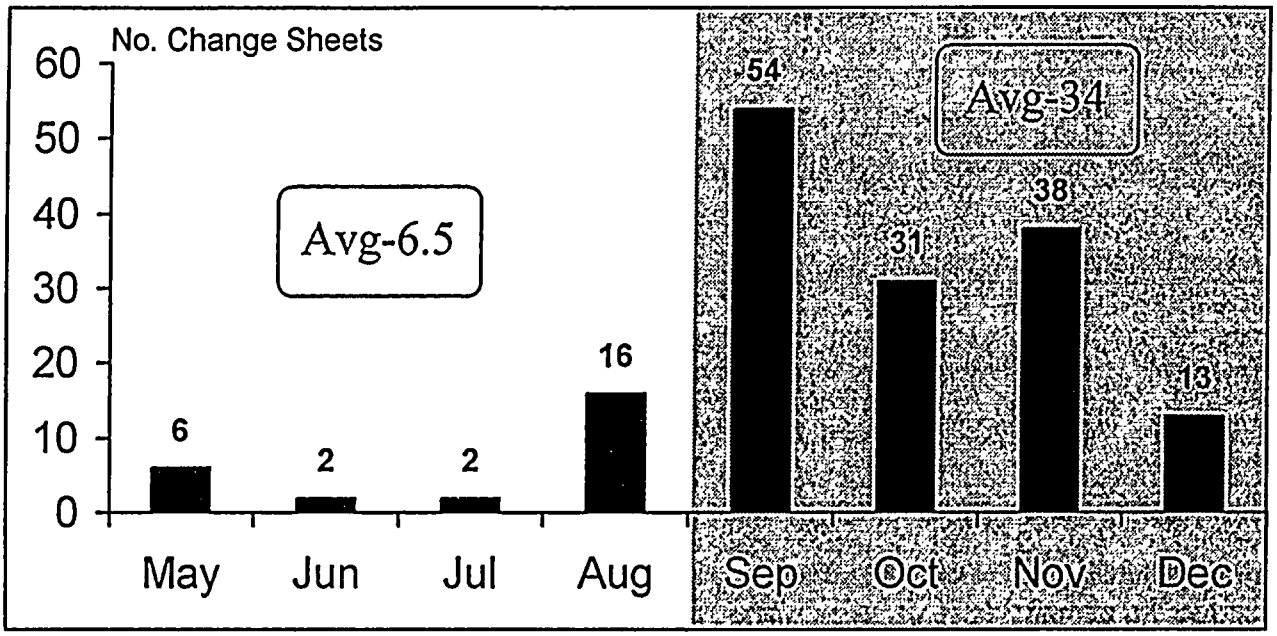
Material Condition

Work Control

"Improvement Culture"

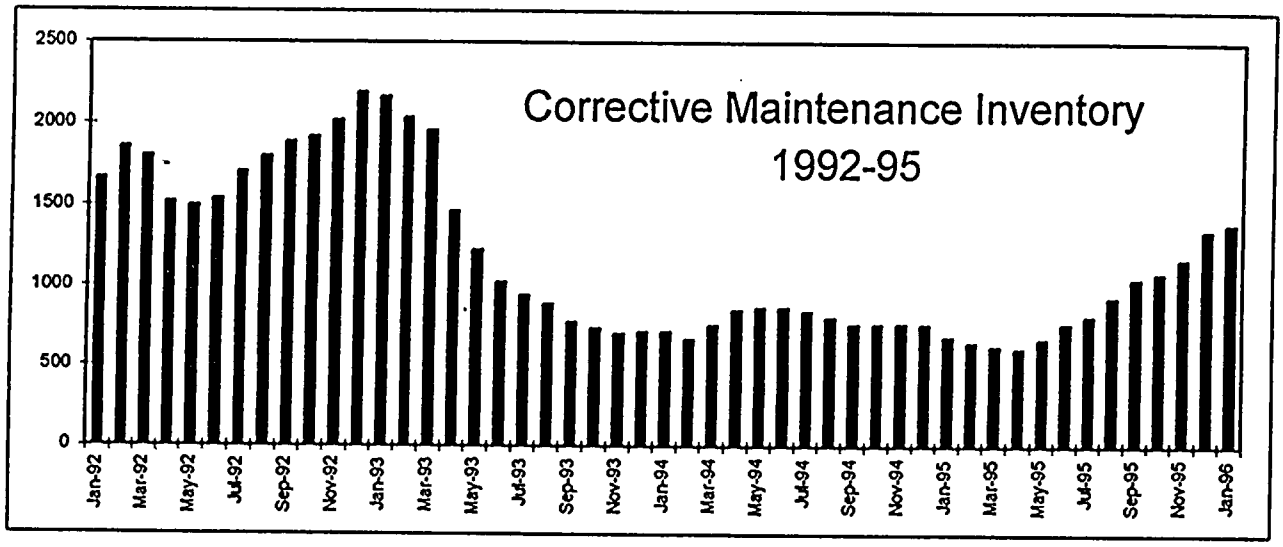
Operations

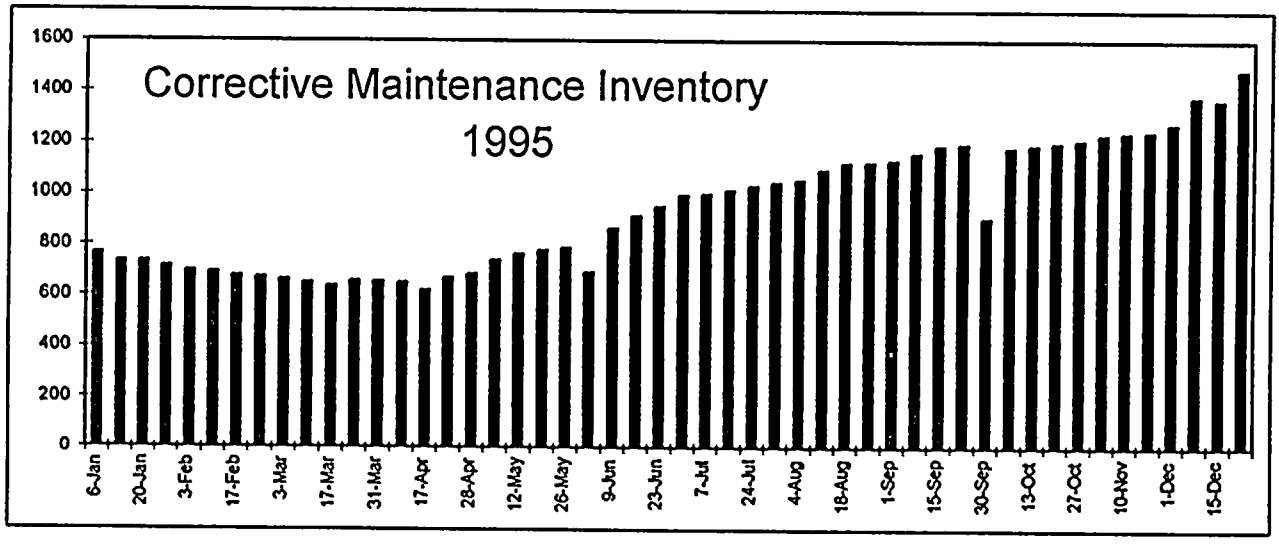
Shifts - Generated Procedure Change Sheets



- Supervisory Oversite
- Procedures
- Briefings/Communication

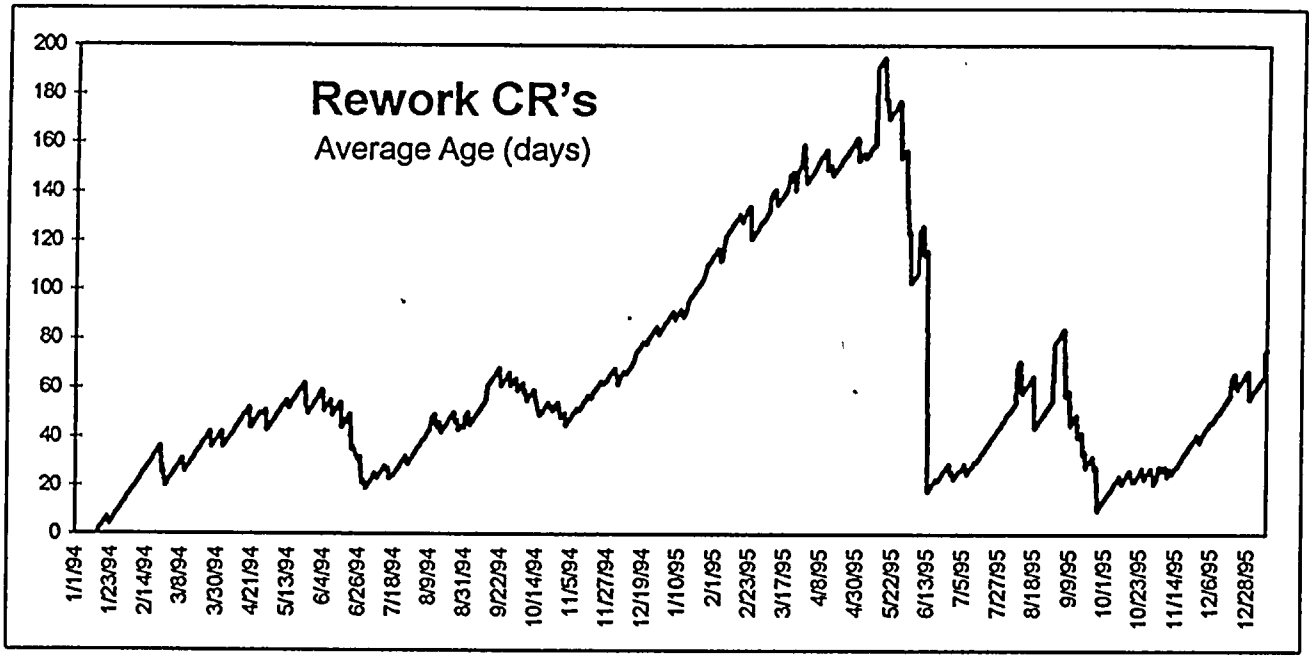
Scheduling





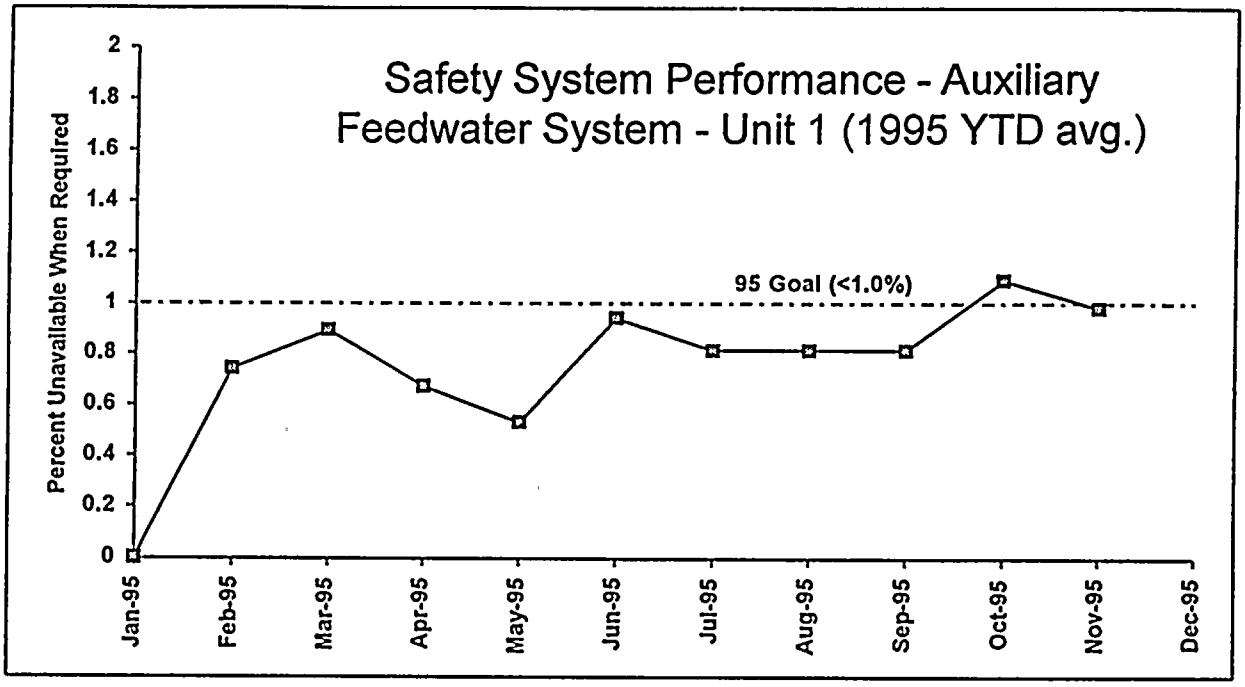
- History Of Work Control Process
- Analysis of Current Status
- Improvement Measures

Maintenance



- Recent Observations
- Quality
  - Corrective Actions
  - Workmanship

Plant Engineering



- Observations and Assessment
- Engineering Support of Production
- Corrective Actions and Self Assessment

Notes