U.S. NUCLEAR REGULATORY COMMISSION

REGION III

REPORT NO. 50-255/95008

FACILITY
Palisades Nuclear Generating Plant

LICENSEE
Palisades Nuclear Generating Plant
27780 Blue Star Memorial Highway
Covert, MI 49043-9530

DATES
May 28 through July 3, 1995.

INSPECTORS
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APPROVED BY

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7/25/95
Date

AREAS INSPECTED

A routine, unannounced inspection of operations, engineering, maintenance, and plant support was performed. Safety assessment and quality verification activities were routinely evaluated.

RESULTS

Assessment of Performance

There was **improvement in coordination and communication of outage activities** among all departments since the beginning of the refueling outage, when a series of errors occurred in a short period of time. These errors were described in the previous inspection report No. 50-255/95007(DRP). Positive observations were observed in planning for major projects, reactor vessel disassembly activities, operator performance of nonroutine activities, motor operated valve testing, and in Engineering Department's implementation of the Alloy 600 project.

The licensee exhibited adequate safety focus during this inspection period. This was evidenced by **good management oversight** of fuel reconstitution activities, the lowering threshold to identify and to correct problems in a timely manner, and in recognizing and planning for potential test problems associated with motor operated valves. However, there were **weaknesses in personnel attention to detail**, a theme observed in the previous inspection report period. These weaknesses included:

- control of dosimetry of personnel entering the radiological controlled area
- mobile crane safety
- inattention that led to an error in setting the reactor vessel head
- control of foreign material within the debris free zones of the reactor cavity and spent fuel pool.

Performance within the area of **OPERATIONS was good** (see Section 1.0). Since the beginning of the refueling outage, improvement was noted in communication, coordination, and management oversight of activities. This was evidenced in the Operations Manager's decision to stop fuel inspections based on the inspection crew becoming too narrowly focused on solving an immediate problem rather than the overall safety picture.

Housekeeping in most areas of the auxiliary building was good. Housekeeping in the east safeguards component cooling water rooms and in the containment was adequate. There were examples of material inappropriately stored on the landings of stairways and weaknesses were noted in the control of foreign material within the debris-free zone near the reactor cavity. Radiological controls and postings in the auxiliary and containment buildings were all within regulatory requirements.

Performance within the area of **MAINTENANCE was adequate** (see Section 2.0). The reactor vessel disassembly activities, including removal of the reactor vessel head, upper guide structure, and core support barrel was conducted well. During these major activities, the pre-job briefing packages were good;

the actual pre-job briefings were thorough and effective. **Management involvement was evident** with standards and expectations being reinforced. Communication and coordination was also good. Workers were cognizant of their respective responsibilities, and personnel access to the work areas was effectively controlled. Lessons learned from previous problems were incorporated and were effective. The licensee's control of contractors was good.

A personnel error occurred during the removal of cables from the reactor vessel head. Maintenance workers mistakenly disconnected four core exit thermocouples (CETs) intended to be used as a backup indication for primary coolant temperature in the event of a loss of shutdown cooling. This event demonstrated **poor coordination and communication** among work groups. However, the licensee's prompt identification and investigation of this error was viewed as **a positive example** of the licensee's lowering threshold to identify potential problems.

Prior to the loading of fuel, a weakness was observed during the interim setting of the reactor vessel head on the reactor vessel. Personnel error on the part of the contractor crew resulted in unexpectedly lowering the head onto the vessel.

Also, a personnel error occurred when a mobile Grove crane was driven to the switchyard to assist in the lift of a battery load bank off the back of a truck. While enroute to the switchyard, the driver of the crane inadvertently severed an overhead 220 VAC power line supplying lighting to the main parking lot and guard house. No personnel injuries were reported, and no other equipment was affected. The cause appeared to be due to inattention to detail. During the previous inspection period, a mobile crane collided with an overhead support structure in the plant protected area. **The inspectors considered the licensee's control of mobile cranes to be a weakness.** This was an Unresolved Item.

Performance within the area of **ENGINEERING was good** (see Section 3.0). Engineering's performance of the Alloy 600 project was good. The licensee's use of mockups to demonstrate and qualify nondestructive examination techniques used in the inspections was a strength, as was the use of "enhanced" ultrasonic inspection techniques. Among weaknesses identified with the plan, the significant reduction in planned inspection scope, precipitated by errors in estimating radiation dose and ALARA concerns, had the largest impact on the plan.

Work to support testing of motor operated valves was performed in a well coordinated manner as evidenced by effective pre-test briefings and good support from operations and contracted technicians. Operations Department's dedicated support in such items as breaker and valve line-ups also contributed to the considerable amount of testing completion. **Engineering Department demonstrated good safety insights to potential test problems.**

Performance within the area of **PLANT SUPPORT was adequate** (see Section 4.0). Activities associated with **removal of the core support barrel and the transfer of the incore detector cask were well planned and executed.** Both activities

demonstrated that lessons learned from difficulties with previous high radiological risk projects (dry cask storage for example) had been applied. However, there were examples where **the licensee was not aggressive at reducing dose** for these and other jobs.

Total outage dose, after completion of approximately 33 percent of the outage scope, was approximately 135 person-rem (1.35 person sievert). This total was approximately 17 percent over the goal for that stage in the outage. If the trend continued, the total outage dose would be approximately 310 person-rem or 24 person- rem over the original goal of 286 person-rem. This overage was due, in large part, to the unexpectedly high doses accrued during the Alloy 600 project. For a multitude of reasons, the plant experienced difficulties in meeting its ALARA goal for the Alloy 600 project.

A noncited violation was issued when an individual entered the radiologically controlled area without an electronic dosimeter. Several other incidents occurred involving the entrance of plant personnel into the radiologically controlled area with dosimetry turned off.

Summary of Open Items

Inspector Follow-up Item: none
Unresolved Item: identified in section 2.2.2
Non-cited Violations: identified in section 4.4

INSPECTION DETAILS

1.0 OPERATIONS

NRC Inspection Procedures 71707 and 92709 were used in the performance of an inspection of ongoing plant operations. The findings showed performance was good.

1.1 Performance of Operations.

Since the beginning of the refueling outage, there was overall improvement in communication, coordination, and management oversight of operations department activities. Operators performed many activities without errors, including surveillances, various system realignments, drain down of the primary coolant system, and core offload. During reactor vessel disassembly activities, **good leadership was demonstrated by shift supervisors.** Operators were aware of the status of equipment important to shutdown safety. In addition, **operators exhibited a good questioning attitude while on rounds.** One example was noted when an auxiliary operator questioned the impact of power lines running through the containment hatch opening on containment closure time.

However, some problems indicated that **improvement in coordination and communication among other groups with plant operators was still warranted.** One example occurred when operators stroked closed containment sump outlet isolation valve CV-3029 while workers were setting up inspection equipment inside the piping near this valve. Management stopped work and regrouped to develop an adequate work plan for the pipe inspections.

1.2 Limiting Conditions For Operation (LCO) Board Annex

The inspectors reviewed the "Limiting Conditions For Operation (LCO) Board Annex" published during the period of June 19 to June 30, 1995. This annex listed the inoperable, non-conforming, or degraded equipment; the condition when the equipment was required to be operable; the applicable work orders or corrective action documents; and the applicable Technical Specification (TS) or administrative requirement. During a review of the annex the **inspectors identified the following errors** pertaining to when the equipment was required to be operable which were discussed with the Operation's Department Superintendent:

- Page 8 of the June 21 LCO Board Annex stated that eight of the steam generator main steam relief valves were removed on June 15. The plant condition was specified as "prior to critical" per TS 3.1.7. Since heatup without the main steam relief installed would have been impractical, the inspectors questioned the appropriateness of this plant condition. The crew referenced the correct LCO, but the inspectors questioned if an integrated system picture had been used when specifying the required plant condition.

On June 23 the inspectors reviewed the controlled copy of the TSs maintained in Document Control Center (DCC) and noted that TS amendment dated June 5 revised TS 3.1.7. This amendment required operability of the main steam reliefs prior to leaving cold shutdown. The amendment was faxed to the site from NRR on June 6, posted in the DCC's TS on June 23, and distributed to the other departments for posting in copies of the TS. The inspectors questioned the appropriateness of the posting delay because, on June 15, the onshift crew made an operability determination for the main steam reliefs using out dated TSs.

The correct plant condition was referenced after discussion with the Operations Department Superintendent.

- Page 11 and 12 of the June 26 LCO Board Annex listed six entries for equipment associated with containment cooling. The equipment included component cooling water heat exchangers, containment air coolers, and containment spray pumps. The required plant condition was specified as "prior to critical" per TS 3.4.1. This was the correct plant condition per TS 3.4.1 but the licensee had implemented a more restrictive operability condition for TS 3.4.1 per standing order (SO) 54. In this case SO 54 required operability of the equipment prior to 300 degrees. SO 54 was not referenced for these items but was correctly referenced in other entries.

- The general listing of components on the LCO Board Annex was hard to follow because they were randomly listed and not listed by in any specific order such as by system, TS, or any other logical order.

Since the plant was in an outage and out of service equipment was controlled by the outage schedule, the items discussed above appeared to have been administrative in nature. However, the inspectors questioned the usefulness of the LCO Board Annex due to the errors. In addition, **checks to assure proper TSs or SOs reference appeared to have been lacking.**

1.3 Licensee Plans For Coping With A Strike

The licensee's company-wide union contract the Utility Workers of America expired at midnight on May 31, 1995. The inspectors reviewed the licensee's strike contingency plan. The plan described measures the licensee would have taken in the event of a strike to maintain satisfactory control over plant activities. **The inspectors determined the plan was adequate.** The inspectors specifically verified that minimum licensed operator staffing would have been maintained in the event of the strike. Prior to the deadline, the licensee and union reached a tentative agreement. The strike contingency was still in effect at the close of this inspection period because the union membership was scheduled to vote on ratification in mid-July.

1.4 Results of Plant Tours

Housekeeping in most areas of the auxiliary building was good. However, **housekeeping weaknesses were noted in the east safeguards and component**

cooling water rooms. Multiple examples of accumulated debris on the floors and pools of water spreading from contaminated drains into adjacent clean areas were noted. In the east safeguards room, the inspectors noted improper securing of an argon gas cylinder to some scaffolding. In the component water cooling room, the inspectors noted the improper securing of ladders and that several workers not wearing hard hats. The inspectors reported these findings to cognizant personnel, and the problems were corrected by the end of the inspection. Radiological controls and postings were all within the regulatory requirements.

Housekeeping in most areas of containment was adequate. Some areas showed specific weaknesses. For example, there were examples of material inappropriately stored on the landings of stairways. An unsecured (not tied off) ladder was propped against the wall adjacent to the reactor vessel water level tygon tube. **There were some weaknesses in the control of foreign material within the debris-free zone near the reactor cavity.** The inspector found broken pieces of glass and broken pieces of signboard within the designated debris-free area. Further, the licensee found multiple examples of unauthorized material in the debris free zones for both the spent fuel and reactor cavity. The licensee was aware of the weakness and initiated condition reports to evaluate the issue. Radiological controls and postings in the containment were all within regulatory requirements.

2.0 MAINTENANCE

NRC Inspection Procedures 62703 and 61726 were used to perform an inspection of maintenance and testing activities. There was one Unresolved Item identified pertaining to use of mobile cranes. The findings showed **maintenance was adequate.**

2.1 Reactor Vessel Disassembly

The inspectors reviewed several activities associated with disassembly of the reactor vessel, performed in accordance with work order 24413169. The licensee performed the activities using procedure C-PAL-RFM-001, "Palisades Refueling Manual."

2.1.1 Palisades Refueling Manual Section 9.2.10, "Head Removal"

The licensee's performance during **lifting and removing of the reactor vessel head was good.** The pre-job briefing was conducted well. The Westinghouse refueling supervisor satisfactorily covered the details, discussed precautions, potentials problems, and contingencies. Management involvement was evident; standards and expectations were reinforced. The actual lift was performed without incident. Communication and coordination were good.

2.1.2 Palisades Refueling Manual Section 9.2.13, "Removal of Upper Guide Structure from Reactor Vessel"

The **removal of upper guide structure** from the reactor vessel was performed well. The activity was well **planned and executed.** Good coordination and communication among the various groups involved was noted. The licensee

demonstrated good contractor control. The licensee performed a thorough search and found no attached fuel assemblies.

2.1.3 Palisades Refueling Manual Section 9.2.16, "Removal and Reinstallation of Core Support Barrel"

Removal of the core barrel was well planned and executed. The briefing package was excellent; the pre-job briefing was thorough and effective in relating radiological controls concerns. The workers were cognizant of their respective responsibilities, and personnel access to the work areas was effectively controlled. Refer to paragraph 4.1.1 for a more thorough discussion of radiological control practices associated with the removal of the core barrel.

2.1.4 Core Barrel and Upper Guide Structure (UGS) Interim installation per paragraphs 9.2.16.D and 9.3 of CPAL RFM 001

The inspectors attended the prejob briefing for both activities. The **briefings were an interactive exchange of information between supervisors and crew members.** Items discussed included applicable procedural steps, critical work positions and who would man them, stopping points, contingency actions, and the radiological work permit.

The inspectors observed a portion of the UGS installation and noted that **several lessons learned from previous UGS removal problems (fuel bundle remained attached and was lifted with the UGS) were incorporated.** For example, the load cell calibration was current, the levelness of the UGS was confirmed, a remote submarine was used to verify cleanliness of the UGS seating flange, and the submarine was used to assure that fuel alignment pins were not damaged during the move. Previously, the fuel alignment pins were gauged to assure straightness. However, during discussion with the engineers, the inspectors were not sure if the pins would be gauged after the UGS was removed from the vessel to facilitate core reload. The gauging was not mandated by regulatory requirements but implemented by the licensee subsequent to the last time a fuel bundle remained attached to the UGS. The licensee was encouraged to evaluate the benefits of the gauging prior to the final setting of UGS.

During the **UGS cleanliness inspection a piece of broken glass was noted on the flange** which was removed. Apparently the glass was the remains of a light bulb that broke during a previous UGS inspection.

2.1.5 Palisades Refueling Manual Section 9.3.2, "Setting of Reactor Head on Shims"

Personnel error on the part of the crew performing this evolution **cost the licensee additional dose** and caused the redirection of resources to perform emergent inspections of the reactor vessel head and upper guide structure components. While lowering the reactor vessel head onto the reactor vessel, the licensee intended to lower the head to about two feet above the reactor vessel flange and to hold for ISI inspections on the flange. **Communication and coordination between personnel in the reactor cavity and the crane**

operator broke down. At this point, the crane operator lowered the reactor vessel head onto the reactor vessel flange prior to the performance of ISI inspections. The licensee has planned various inspections and tests to check for damage to susceptible components.

2.2. Other Maintenance Observations

2.2.1 Work Order 24415662, Removal Of Cables From The Reactor Vessel Head

This work activity was another example of poor coordination and communication among work groups. This example occurred early in the refueling outage and was discussed in the previous inspection report (IR95007(DRP)). Maintenance workers mistakenly disconnected four core exit thermocouples intended to be used as one backup indication for primary coolant temperature in the event of a loss of shutdown cooling. The maintenance worker involved recognized and immediately reported his mistake to his supervisor and to plant operators. The core exit thermocouples were not required to be operable per the Technical Specifications and were reconnected shortly thereafter. The licensee held an immediate corrective action review board with all involved personnel to discuss causes and corrective actions.

2.2.2 Work Order 24416029, Capacity Test Of Switch Yard Battery

A personnel error associated with this activity occurred when a mobile Grove crane was driven to the switchyard to assist in the lift of a battery load bank off the back of a truck. The crane driver inadvertently severed an overhead power line supplying lighting to the main parking lot and guard house. No personnel injuries were reported, and no other equipment was reported to have been affected. The cause appeared to be due to inattention to detail. Because a similar event had occurred during the previous inspection period when a mobile crane collided with an overhead support structure, the inspectors considered the licensee's control of mobile cranes to be a weakness. The licensee took immediate action to investigate and to correct the cause of this recent event. The licensee agreed to respond within 60 days on causes and preventive actions for this event. Pending review of the licensee's response, this was considered an unresolved item (50-255/95008-01).

Activities associated with work order 24416287, Cleaning Of 1-2 DG Jacket Water And Lube Oil Coolers was observed with no concerns being identified.

3.0 ENGINEERING

NRC Inspection Procedures 37551, 73051, 73052, and 73755 were used to perform an inspection of engineering activities. The findings showed performance was good. Items which were "Closed" as a result of this inspection met the criteria established in the Inspection Procedures.

3.1 Inservice Inspection (Inconel Alloy 600 components)

The licensee's Project Plan Alloy 600, Revision 1, represented a sound technical approach to managing primary water stress corrosion cracking (PWSCC)

in Inconel Alloy 600 materials within the primary coolant system (PCS). The nondestructive testing (NDE) **techniques and acceptance criteria used were reasonable and consistent with analysis and industry practices.** The NDE inspections completed this outage, albeit reduced in scope, still assured components in the most susceptible category, Group I, and the more susceptible components category, Group II, received ultrasonic (UT) or dye penetrant (PT) and visual inspection (VT). All identified Alloy 600 locations in the PCS received a visual inspection as a minimum. Moreover, **no PWSCC was found during inspections** performed this outage.

3.1.1 Program Review

The Alloy 600 plan included as one of several goals, the development of an inspection program to identify and to characterize PWSCC in Inconel Alloy 600 components. Toward this goal, the plan identified 251 Inconel Alloy components potentially susceptible to PWSCC in the plant. These components were grouped into three levels of inspection priority based on susceptibility to PWSCC, consequence of failure, detectability and ALARA considerations. The **original scope** of planned inspections was **subsequently reduced to save total project radiation dose expenditure**. Of the original 79 UT inspections planned, 27 were completed. Of the original 66 PT inspections planned, 61 were completed. The completed inspections covered all Inconel Alloy 600 components listed as the highest priority (group I) in the plan. The licensee had not committed to perform inspections deleted for this outage in the lower priority inspection categories listed in the original plan.

The following observations were noted as **weaknesses associated with the planned inspection** of Inconel Alloy 600 components:

- Previously unidentified flow diverter plate in the pressurizer spray line prevented planned visual inspection of the internal weld surfaces on the pressurizer spray safe end;

- The use of PT inspections on outside surfaces of components to locate PWSCC, which would initiate from the interior surfaces of components, was at best redundant with UT or VT performed. Dose expended on these PT inspections was not considered consistent with ALARA principles;

- The underestimate of radiation dose expenditure associated with planned inspection activities, precipitated the significant scope reduction in actual completed inspection activities.

The following observations were noted as **strengths associated with the planned inspection** of Inconel Alloy 600 components:

- The utilization of mockups to demonstrate and to qualify NDE techniques used in the inspections;

- The use of "enhanced" UT techniques, which included angle beam (shear and longitudinal wave mode) search units, automated data collection and

analysis and scanning equipment custom designed and built specifically for Palisades components.

3.1.2 Observation of Work Activities

Personnel from B&W Nuclear Technologies (BWNT) performed the inservice inspection (ISI) of Inconel Alloy 600 components in accordance with Palisades' Alloy 600 Project Plan and Alloy 600 specification revision 0. The NRC inspectors observed work activities and had discussions with BWNT and licensee personnel during ISI activities. These observations included:

- BWNT personnel performing UT on pressurizer spray line safe end and shutdown cooling outlet connection safe end welds;
- BWNT personnel performing PT on the pressurizer surge line safe end welds, hot leg loop pressure piping safe end welds and hot leg drain line safe end welds;
- BWNT personnel performing UT on the replacement power operated relief valve (PORV) safe end upper weld.

3.1.3 Procedure Review

ISI procedures used in support of the Alloy 600 Project Plan were reviewed by the NRC inspectors. The ISI procedures were found to be acceptable and in accordance with ASME section V, 1989 edition as modified by requirements of the Alloy 600 Inspection Specification revision 0. The inspectors reviewed qualifications and certifications of all BWNT personnel performing ISI, verifying conformance with licensees' Alloy 600 specification revision 0 requirements.

3.1.4 Data Review

The UT and PT examination data reviewed was found to be in accordance with the applicable ISI procedures and the Alloy 600 Specification, Revision 0. Radiographs of the upper PORV safe end weld were reviewed by the inspectors and found to be in conformance to ASME section V, 1986 addenda requirements. Radiographic films taken for the lower PORV safe end weld could not meet code requirements for geometric unsharpness and 2T hole penetrameter image quality requirements. The licensee subsequently performed ultrasonic inspection of this weld to meet ASME section III, 1986 addenda requirements for volumetric examination. No reportable indications were found for the UT inspections performed or for the radiographic inspections of the upper PORV safe end weld. Four components had reportable surface indications disclosed through PT testing. These indications were subsequently removed and follow up PTs verified no further surface indications to be present.

3.2 MOV Testing to Support GL 89-10 Commitments

Work to support testing of MOVs was performed in a well coordinated manner due to effective pre-test briefings and good support from operations and

contracted technicians. Operations Department's dedicated support in such items as breaker and valve line-ups also contributed to the considerable amount of testing completion.

Engineering Department demonstrated good safety insights to potential test problems. For example, prior to performing test T-352, "HPSI Loop Isolation MOVs MO-3007, 3009, 3011, 3013 Differential Pressure Test," the test coordinator anticipated a potential overthrust condition for MO-3009 because of the flow-over-the-seat valve design. The valve vendor was contacted and indicated that the valve weak link thrust limit of 15,151 pounds was conservative and a thrust up to 20,000 pounds for several hundred cycles was acceptable. During actual testing MO-3009 experienced approx 13,181 pounds of thrust.

3.2.1 Fuel Inspection and Reconstitution per I-FC-942-01

The licensee performed an inspection of the fuel bundles that will be reused during the next fuel cycle. The purpose of this inspection was to identity any cladding failures and to determine if there were any grid strap relaxation problems with fuel bundles used for vessel beltline weld shielding. Cladding integrity was confirmed by ultra-sonic inspection. Grid strap tension measurements required removal of the top plate and pull test on selected pins.

During the grid strap inspection, the **operations manager observed the crew's recovery from a stuck tool**. The manager alertly observed that a "red" mark located on the tool almost came out of the water. This mark was used to assure that sufficient water shielding was maintained between the crew and the attached component. Apparently, the **crew was narrowly focused** on the recovery activities. It was watching the underwater camera and not monitoring the "big picture". The **manager's observation prompted a shutdown of this activity until the crew was briefed on the procedure and equipment.**

While observing the ultra-sonic inspection, the inspectors discussed the scope of the inspection with the fuel engineer. The scope included approximately 330 bundles. This number included the bundles scheduled to be reused for the next fuel cycle and those selected for dry cask storage. During this conservation the inspectors was informed that the licensee's practice has been to confirm the integrity of the fuel bundles selected for dry cask storage by performing the mandated visual inspection and additional inspections such as ultra-sonic or sipping to confirm the integrity of the fuel pin cladding.

3.2.2 Service Water Enhancement per I-FC-959-01

The enhancement required isolation of cooling water to the spent fuel pool (SFP) heat exchangers. To provide SFP cooling, a temporary cooling system was installed. The inspectors reviewed the installation and operation procedures. Once the system was installed, the inspectors observed system performance and confirmed that adequate SFP cooling was available and maintained.

The **installation and operating procedures addressed many contingencies**, provided minimum operating temperatures and maximum SFP temperatures, specified when and how normal SFP cooling shall be reestablished, and ensured

equipment reliability of the temporary equipment through pre-certification operability runs. When the system was placed in service, SFP temperature remained essentially stable.

3.3 Follow-up on Non-Routine Events.

NRC Inspection Procedure 92904 was used to perform a review of written reports on non-routine events. The following item was closed:

(Closed) LER 50-255/93009: Through wall cracking, caused by primary water stress corrosion cracking (PWSCC) in the heat affected zone of the PORV line to pressurizer Inconel Alloy 600 safe-end weld. The cracking was circumferential and initiated from the inside diameter progressing intergranularly with the final 5-10 percent of crack growth being transgranular.

To prevent reoccurrence of this event, the licensee replaced the Inconel Alloy 600 pressurizer PORV safe-end with a type 316L stainless steel safe-end and used mechanical stress improvement techniques on select components to reduce susceptibility to PWSCC. In addition the licensee implemented an inspection of susceptible Alloy 600 components in the plant and performed a fracture mechanics analysis for Alloy 600 components to demonstrate that they would fulfill their inservice lifetimes. This item is closed.

3.4 Follow-up on Previously Opened Items.

NRC Inspection Procedure 92904 was used to perform a review of previously opened items (violations, unresolved items, and inspection follow-up items). No problems were identified, and the following item was closed:

(Closed) Violation (50-255/94015-01): Failure to take prompt and adequate corrective actions for the review of a 10 CFR Part 21 notice, and the evaluation of completed MOV tests, which included overthrust and overtorque conditions. In response, the licensee developed an MOV Program Recovery Plan that established MOV program directives, provided proper operability acceptance criteria for MOV test procedures, and prioritized available resources. In addition, the licensee revised engineering manual procedure EM-28-01, "Motor Operated Valve Program," to clarify the evaluation of completed test data. The 10 CFR Part 21 notice was evaluated and results were incorporated in the MOV torque/thrust calculations. This item is closed.

4.0 PLANT SUPPORT (IPs 71750 and 83750)

Total outage dose, was approximately 135 rem (1.35 Sievert (Sv)), with about 33 percent of the outage scope completed. This total exceeded the dose goal by about 17 percent, largely owing to difficulties with the Alloy 600 project (section 4.2). Several poor radworker practices were identified during reviews of outage work (section 4.1), and a non-cited violation was issued for inadequate personnel monitoring (section 4.3). Overall, the radiation protection (RP) department performance was considered adequate.

4.1 Removal of the Core Support Barrel and the Transfer of the Incore Detector Cask

The inspectors observed two radiologically significant outage activities; the removal of the core support barrel and the transfer of the incore detector cask. Although both projects used "lessons learned" from similar work, several concerns were identified regarding radworker practices.

A lead shield was used to lower dose rates around the barrel to around 60-100 mrem (0.6-1.0 mSv) per hour (from > 1 rem (10 mSv) per hour). During the moving of the barrel, several workers were observed loitering around the barrel, and the RP technician was moving frequently from behind the shield to inspect the barrel. Similar observations were made during the incore cask removal, where workers were observed loitering in area dose rates between 5-20 mrem (0.05 and 0.20 mSv) per hour. In neither case (core barrel or incore cask) were the workers' behavior challenged, suggesting a lack of aggressiveness by the licensee in reducing individual dose.

4.2 Alloy 600 Project

The initial scope of the Alloy 600 project involved the examination of 79 primary loop penetrations using dye-penetrant (PT) and ultrasound testing (UT) in an effort to detect primary water stress corrosion cracking (PWSCC) on the interior of the penetrations. Based on the vendor's estimate of the time needed to perform the examinations using remote UT equipment, the plant developed an ALARA goal of 10 person-rem. However, from the beginning of the work, the **plant experienced difficulties in meeting its ALARA goal for this job.**

Due to clearance constraints, the vendor was unable to use remotely operated automatic UT equipment, and had to rely on manually operated equipment that required the operator to be in close proximity to the piping being examined. Radiation levels near the piping were much higher than the levels expected for the use of remote equipment. The higher radiation levels, thus, increased the committed dose for the project.

Aside from the inability to use remotely operated UT equipment, the **erroneous use of the vendor's time estimate** to complete the Alloy 600 ISI examinations **made the largest impact on the licensee's ability to establish an accurate ALARA estimate.** The time estimate provided by the vendor was in clock-hours, rather than person-hours. This significant difference in time was not effectively communicated between the licensee and the vendor. Although the vendor's "clock-hour" estimate was a factor of three different from the licensee's estimate of the time required to complete the examinations, the licensee did not question the reasoning behind the difference. The licensee accepted the vendor's time estimate, and believing the time was expressed in person-hours, used that time in developing the ALARA projection for the work package.

The licensee documented the UT equipment and time estimate problems in an In-progress Review report, and indicated that long term corrective actions and lessons learned would be included in a post job review planned for after the outage.

4.3 External Exposure Control

On June 20, 1995, an individual violated station procedure by entering the radiologically controlled area (RCA) without an electronic dosimeter (ED). A review of selected licensee Condition Reports identified several other incidents involving EDs set on "pause" (i.e. not logged into the computer). Station Procedure Number 7.04, revision 14, states, in part, that individuals shall log in on the Management Information System (MIS) with a Secondary Dosimeter (ED) prior to entering the RCA. The procedure would have been violated if an individual entered the RCA with an ED that was improperly set on "pause" or an individual entered the RCA without an ED. For each event, the workers were counseled by RP personnel and the respective supervisors were notified. The access of the worker involved in the June 20th incident was revoked for an unspecified amount of time. The licensee was developing a formal process for taking action against individuals who violate RP procedures and practices. Although these events, in the aggregate, were a violation of the above station procedure, it was identified by the licensee and immediate corrective actions were taken. Therefore, it will not be cited as the criteria specified in Section VII.B.2 of the "General Statement of Policy and Procedures for NRC Enforcement Actions", (Enforcement Policy, 10 CFR Part 2, Appendix C), were met.

5.0 PERSONS CONTACTED AND MANAGEMENT MEETINGS

The inspectors 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 July 6, 1995, the inspectors 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 inspectors are proprietary.

*R. A. Fenech, Vice President, Nuclear Operations
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*R. B. Kasper, Maintenance Manager
R. C. Miller, Deputy Engineering and Modifications Manager
*C. R. Ritt, Administrative Manager
R. M. Rice, System Engineering Manager