U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No.:	50-255
License No.:	DPR-20
Report No.:	50-255/97008(DRP)
Licensee:	Consumers Power Company 212 West Michigan Avenue Jackson, MI 49201
Facility:	Palisades Nuclear Generating Plant
Location:	27780 Blue Star Memorial Highway Covert, MI 49043-9530
Dates:	May 24 through July 7, 1997
Inspector:	P. Prescott, Resident Inspector
Approved by:	Bruce L. Burgess, Chief Reactor Projects Branch 6

EXECUTIVE SUMMARY

Palisades Nuclear Generating Plant

NRC Inspection Report 50-255/97008

This inspection reviewed aspects of licensee operations, maintenance, engineering and plant support. The report covers a 6-week period of resident inspection.

Operations

- A plant procedure that allowed operations with steady state indicated reactor thermal power greater than the licensed limit was identified as a violation. This procedure had previously been modified by the licensee to no longer allow the steady state operation above the licensed limit. While no actual operation of the unit of greater than the licensed limit was identified, the potential for such operation had existed. (Section 01.2)

- The inspectors noted good operations performance during a CCW system spurious relief valve lift. Operator identification and resolution of the event was prompt and thorough. However, the inspectors identified a weakness in the initial operability evaluation, which was subsequently addressed. The relief valve was subsequently gagged closed until repairs could be initiated. (Section 01.3)

Maintenance

- The inspectors observed a weakness in communications in that neither system engineering nor I&C personnel informed the operators of a grounding problem that could occur during performance of the loop one T-ref maintenance activity, nor were the alarms that could be received in the control room reviewed with the operators. The inspectors noted these oversights were corrected in the loop two phase of the maintenance activity. (Section M1.2)

- The inspector discussed an improper post maintenance test on valve CV-0733, and indicated that this was another example of a negative trend observed in the quality of post maintenance testing. The PMTs reviewed appeared to have been written to verify the initial problem was repaired, **not** that the component continued to meet its design function following maintenance. The licensee is currently reviewing the PMT process. (Section M1.3)

Engineering

- The inspectors, in followup to a potentially generic issue, determined that the licensee's administrative and design features that pertained to part length (P-L) control rods provide sufficient controls such that a reactor power excursion due to a stuck or mispositioned P-L control rod would be highly unlikely. Also, the licensee's fuel vendor had reviewed and determined that a P-L control rod event was bounded by a dropped or ejected control rod scenario in the current fuel cycle analysis report. (Section E1.1)

Plant Support

- The inspectors determined that the post maintenance critique did not fully address other available options to reduce dose during evaporation cleaning activities. Critique meeting participants characterized the evaporator cleaning as a low dose job (less than or equal to 10 mrem) when in fact the licensee had expended approximately 350 mrem for a job that may not have been required. The inspectors concluded that the evaporator cleaning job did not have the proper emphasis placed on ALARA planning. (Section R1.1)

REPORT DETAILS

Summary of Plant Status

The plant operated at essentially 99.6 percent power for the entire inspection report period. July 7, 1997, marked the 138th day of continuous power operation.

I. Operations

01 Conduct of Operations

01.1 General Comments (71707)

Using Inspection Procedure 71707, the inspectors conducted frequent reviews of ongoing plant operations. The inspectors considered the conduct of operations to be good. Specific events and noteworthy observations are detailed below.

01.2 Followup on Exceeding Licensed Thermal Power Limits

a. Inspection Scope (71707)

During this inspection period, the NRC completed its review of enforcement action (EA) 96-092 concerning a February 7, 1996 event at Palisades involving the potential to exceed rated reactor thermal power limits as indicated by available control room power monitors. Inspection report 50-255/96002(DRP) provided the specific facts and preliminary analysis of this event. Below is a discussion of the NRC's review and conclusions concerning the licensee's operation at near full power.

b. Observations and Findings

On February 7, 1996, reactor thermal power was indicated to have exceeded the power stated in the facility's license. This inadvertently occurred during a delithiation evolution to control primary coolant system chemistry parameters. The operations shift was aware that, by procedure GOP-12, Revision 12, reactor power was allowed to reach 100.99 percent. Reactor thermal power is measured by nuclear instrumentation that is calibrated periodically using a heat balance calculation. A heat balance calculation provides the best indication of actual reactor thermal power. Accident analyses presented in the FSAR must meet the requirements of 10 CFR 50 Appendix K "ECCS Evaluation Models." These analyses are performed assuming a reactor thermal power of 102 percent in order to allow for instrument uncertainties. By exceeding licensed thermal power limits, reactor power during an accident scenario could potentially be outside design bases because the margin of safety derived from assuming a 2 percent instrument error would be reduced by the higher initial power level at the time of accident initiation.

However, the inspectors determined the safety significance of this event was minimal. Review of subsequent tests and analyses showed that the licensee did not exceed 100 percent power during the nine hour delithiation process. A calorimetric uncertainty analysis was completed that utilized instrumentation and indication uncertainties and an ultrasonic flow measurement (UFM) of the feedwater flow rate was performed. The UFM provided a more accurate indication of actual feedwater flow, independent of the installed feedwater venturies. Due to feedwater flow rates being the single largest contributor in a calorimetric calculation; small errors in feedwater flow rates could result in larger differences in indicated reactor power. Results of the UFM testing revealed that actual power was 2.2 percent less than the indicated power based on use of the feedwater flow venturies. The difference was due to a conservative initial venturi calibration and venturi fouling. Using the UFM results, maximum power level achieved during the delithiation process was determined to be 98.2 percent.

NRC has issued guidance that licensees may not operate above the steady state indicated reactor thermal power limits stated in the license, except in unanticipated transient conditions. If steady state indicated reactor thermal power exceeds the licensed limit, the guidance directed licensees to initiate prompt corrective action within 15 minutes to restore reactor power to less than or equal to the license power limit.

The inspectors wrote a task interface agreement (TIA) issued to the Office of Nuclear Reactor Regulation (NRR) to evaluate the adequacy of existing guidance. The basis for the TIA was that present technology allows calculating, almost instantaneously, reactor thermal power. The current standing guidance to the industry was developed when calculating reactor thermal power was a one hour or longer process. Thus under the old technology, the delithiation process and resultant indicated power level of over 100 percent would not have been immediately detected. Using current technology, almost anytime any evolution raises power above 100 percent, the power excursion would be detected and raise a question regarding whether or not a licensee should perform a calorimetric knowing an overpower indication exists.

The response to the TIA stated that the deliberate raising of power above the licensed limit was inappropriate. Procedure GOP-12, Revision 12, allowed the brief operation in excess of licensed reactor thermal power. This procedure was inappropriate to the circumstances and is considered a violation of 10 CFR 50 Appendix B, Criterion V "Instructions, Procedures, and Drawings," (50-255/97008-01(DRP)).

In response to NRC concerns, licensee management modified the procedure such that it no longer allowed steady state power operation above the licensed limit.

c. <u>Conclusions</u>

A plant procedure that allowed operations with steady state indicated reactor thermal power greater than the licensed limit was identified as a violation. In response to concerns from the NRC, licensee management modified the procedure. While actual operation of the unit greater than the licensed limit was not identified, the potential for such operation had existed.

01.3 Component Cooling Water (CCW) Relief Valve (RV) Lift During Surveillance

a. Inspection Scope (71707, 61726 and 37551)

The inspectors observed operations personnel conduct a prejob brief and perform a right channel surveillance using procedure QO-1, "Safety Injection System."

b. Observations and Findings

The purpose of surveillance procedure QO-1 was to demonstrate operability of the right channel of the safety injection system (SIS) initiation circuitry (SIS actuation relays and design basis accident (DBA) sequencer) by using the internal testing capability of the system. One system tested is the component cooling water (CCW) system. The SIS initiation circuitry signals one of the other two CCW pumps to start (one normally is already in service). During performance of QO-1 on June 9, 1997, CCW pump P-52B automatically started as required. This resulted in an expected increase in CCW system pressure. However, relief valve RV-2108, which provides thermal over pressure protection for the shield cooling heat exchanger, subsequently lifted.

The valve did not reseat normally which resulted in an approximately two gpm leak. No alarms are automatically actuated when relief valve RV-2108 lifts, thus the operating crew was not immediately aware of the partially open valve. An extra nuclear shift operator (NSO) was assigned to assist in the control room while the two normal onshift NSOs performed QO-1. During a routine panel walkdown, the extra NSO noticed a decrease of approximately 10 percent in the CCW surge tank level. The extra NSO also noted to the control room supervisor that containment sump level was trending up. The operators checked the volume control tank level to verify there was no decrease in level and to ensure that a primary coolant system leak had not occurred. The operators then concentrated on finding a CCW leak.

The operators:

- Calculated CCW surge tank level loss to determine the rate of decrease;
- stopped testing of QO-1;
- restored the plant to normal configuration following the suspension of surveillance test QO-1; and
- entered the off-normal procedure for the CCW system due to the apparent leak.

The off-normal procedure was reviewed by operations personnel and the location of all relief valves on a CCW system drawing were identified. Also, personnel on a standby list of maintenance and system engineering personnel were notified. A CCW corrective action team entered containment and identified that RV-2108 for the shield cooling system had lifted and stayed opened. The valve was mechanically agitated and it subsequently reseated. Licensee personnel generated a condition report and an initial operability evaluation was performed. Operators noted that the available indicators for the relief valve indicated that the valve lifted early since when the relief valve lifted, CCW pressure was approximately 135 psig and the setpoint of the relief valve was 150 psig.

The inspectors noted good operator response to the stuck open relief valve and small CCW leak inside of containment.

The inspectors identified one weakness with the initial operability evaluation. Initially, the evaluation addressed only the as found leak rate of 2 gpm and failed to address the potential leak rate of a full open relief valve. If RV-2108 had lifted to its full capacity of 24 gpm, the inspectors were concerned that the CCW surge tank makeup capability would be insufficient. System engineering calculated that the makeup capability of the CCW system was 150 gpm, which would be sufficient to maintain the CCW system operable should RV-2108 spuriously lift again. Subsequently, the valve was gagged closed to prevent recurrence. Two other relief valves associated with the CCW system were verified to provide adequate protection for the shield cooling heat exchanger from over pressure until RV-2108 can be replaced.

c. Conclusions

The inspectors noted good operator performance during identification and response to the spurious lift of a CCW relief valve. Operator identification and response to restore CCW system integrity was prompt and thorough. However, the inspectors identified a weakness in the initial operability evaluation, which was subsequently addressed. The relief valve was subsequently gagged closed until repairs can be initiated.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

a. Inspection Scope (62707 and 61726)

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

Work Order No:

- 24711110 Dirty waste "B" evaporator; open/inspect and hydrolaze

•	24711266	CV-3223, SDC HXH E-60A inlet valve; open/inspect PCV and replace internals
•	24711268	CV-3212, SDC HXH E-60B inlet valve; open/inspect PCV and replace internals
•	24711416	CV-3055, inlet valve to SDC HXH; open/inspect PCV and replace internals
•	24711267	CV-3224, SDC HXH E-60A outlet valve; open/inspect PCV and replace internals
•	24514371	Install new program for PCS Loop one revised T-ref curve in transmitter TYT-0100 per SC-95-099
•	24612597	Install new program for LIC-010A pressurizer level controller for revised T-ref curve on loop one
•	24612508	CV-0511 turbine bypass valve; replace tubing and fittings downstream of CA-0390
•	24513316	Diagnostic testing of CV-0511
•	24514370	Install new program for PCS loop 2 revised T-ref curve in transmitter TYT-0200 per SC-95-099
•	24612596	Install new program for LIC-0101B pressurizer level controller for revised T-ref curve on loop two
•	24612911	Charging pump P-55A; install new pump body and head
•	24712354	Hydrolaze drain line to equipment drain tank T-80

<u>Surveillance Activities</u>

•	SOP-2	Surveillance for Auxiliary Feedwater valves CV-0727 and CV-0749 following PPAC FWS034
•	SOP-8 ATT 2	Testing of Main Turbine Valves/Protective Trips
•	QO-1	Safety Injection System (Right Channel With Standby Power)
•	QO-1	Safety Injection System (Right Channel Without Standby Power)
•	QO-19	Inservice Test Procedure - HPSI Pump and ESS Check Valve Operability Test

b. Observations and Findings

The inspectors concluded that the work performed during maintenance and surveillance activities was professional and thorough. All work observed was performed with the work package present and in active use. Work packages were comprehensive for the task and post maintenance testing requirements were adequate. The inspectors frequently observed supervisors and system engineers monitoring work practices. When applicable, work was completed by adhering to the appropriate radiation control practices.

c. Conclusions

In general, the inspectors observed good procedure adherence, maintenance and radiation worker practices. Specific observations are detailed below.

M1.2 Poor Communications During T-ref Controller Maintenance

a. Inspection Scope (61726 and 71707)

The Inspectors observed portions of scheduled maintenance on transmitters TYT-0100 and TYT-0200. The temperature reference (T-ref) curve had changed and the licensee intended to change the electronic program constants to reflect the revised curve. In addition to observing the transmitter work, the inspectors also reviewed the associated work package and observed the post maintenance test. Also observed were maintenance activities for the pressurizer level controller LIC-0101A and LIC-0101B, which provided a revised pressurizer level setpoint curve. The pressurizer level setpoint curve was revised to reflect a revised T_{ave} for 100 percent power.

b. Observation and Findings

As noted in section O1.2 of this report, the licensee had identified conservative errors in the measured flow rates of the main feedwater system. Following the identification of these errors, I&C personnel adjusted feedwater flow instrumentation and other power measuring instruments. As the unit power was adjusted, T_{ave} and T-ref were also adjusted.

The first portion of the maintenance activity involved removal of the loop one transmitter TYT-0100 to have its program upgraded and then reinstalled after testing. TYT-0100 was unplugged from the control room panel and a digital programmer was connected. When the programmer was turned on and TYT-0100 was plugged back in an AC ground fault alarm occurred on preferred AC power bus Y-10, which powers TYT-0100. The TYT-0100 showed no sign of having AC power applied. Also, digital T_{ave} indicator TI-0111 and temperature recorder TR-021 both showed a 15°F increase. At this point, the control room operators suspended the job and entered the proper annunciator response procedure. The inspectors observed good command and control of control room operations.

The ground fault was evaluated and the required procedural actions completed. The operators then allowed removal of transmitter TYT-0100. The ground was no longer observed on the Y-10 bus. The original TYT-0100 transmitter unit was replaced with a new unit. The inspectors learned from discussions with the system engineer that similar events had occurred with the same model transmitters in five previous instances. The inspectors had attended the prejob brief and this potential problem was not discussed. The inspectors also noted that the operations personnel were not present for the prejob brief. Prior to commencing work, neither the system engineer nor instrumentation and control (I&C) technicians briefed operations of this potential problem. During this evolution, the inspectors discussed with plant management concerns that operators are briefed on expected alarms prior to commencement of work.

Prior to work on the second T_{ave} loop, the inspectors discussed with operations that LIC-0101B was a suspect unit and that the scope of the job was to only reprogram the unit. The operators performed a prejob brief for the loop two work activity with the operations shift, I&C technicians, their supervisor, and the system engineer. The inspectors noted the brief was thorough. During the brief, the system engineer identified to operations that LIC-0101B was a suspect unit. The original scope of the work package was to simply reprogram the unit and not replace the unit or the power supply. The operators suggested that it would be prudent to take care of the potential power supply problem now rather than simply reinstall the unit. The system engineer agreed and the power supply was replaced after proper work order revisions were completed.

c. Conclusions

The inspectors observed that neither system engineering nor I&C personnel informed the operators of a potential problem that could occur during performance of the T-ref maintenance activity, nor were potential alarms reviewed with operations. The inspectors noted these oversights were corrected in the second phase of the maintenance activity.

M1.3 Adequacy of Post Maintenance Test (PMT) Requirements

a. Inspection Scope 62707

The inspectors observed portions of maintenance performed for turbine bypass valve CV-0511 and portions of the testing conducted on condensate fast makeup valve, CV-0733. The PMT history for the CV-0733 valve was also reviewed.

b. Observations and Findings

The intent of the work order for CV-0511 was to replace a mix of copper and stainless steel instrument air lines and fittings with new stainless steel. Part of the work order required removal of certain solenoid valve (SVs). The SVs were to be de-terminated and the wire-nutted connections replaced with lugged connections.

In the inspectors' review of the PMT operability requirements for CV-0511, CV-0511 was to remain isolated from the main steam system during valve timing tests. Also, the PMT required the verification of no air leakage on the replaced instrument air lines. The inspectors were concerned this would be an inadequate PMT of CV-0511, in that all the SVs would not be verified as functional. The SVs for the bypass valve quick opening and loss of condenser vacuum functions would not be tested. The turbine bypass valve is important to plant safety. The valve passes up to 4.5 percent steam flow with the reactor at full power. The FSAR states that the turbine bypass valve is one of the systems utilized for taking the plant to hot shutdown. The valve is also discussed in the Technical Specification Basis Section 2.2. The TS basis states that additional assurance is provided by the bypass valve in preventing the nuclear steam supply system pressure from exceeding safety limits.

The inspectors observed the PMT for the condensate fast makeup valve. The 12 inch valve supplies condensate makeup from storage tank T-2. The valve opens on a low-low hotwell level signal. A pressure control valve (PCV) was repaired and operators attempted to stroke CV-0733. The valve failed to stroke. The PCV was adjusted to increase air pressure and the valve was mechanically agitated. However, the valve still failed to stroke. The valve maintenance supervisor stopped further testing until the situation could be reviewed. At the end of the inspection period, the valve had not been stroked. Licensee maintenance personnel are evaluating options to stroke the valve.

The licensee, in reviewing post maintenance and interviews with personnel, found that the actuator for CV-0733 had been overhauled in February 1997. After completion of the overhaul, it was found that CV-0733 had not been stroked even though the PMT cover sheet recommended it. The licensee wrote a condition report which requires a root cause analysis (referred to as a level two condition report).

In the previous inspection report (IR) (50-255/97006(DRP)), the inspectors identified a concern with PMT of the P-55A charging pump. Also, detailed in the same IR were problems of PMT with P-8B Auxiliary feedwater pump. The inspectors discussed with the licensee the continued weaknesses noted in the area of post maintenance testing.

c. Conclusions

The licensee reviewed the test requirements and decided to stroke the valve and declare CV-0511 inoperable but available pending testing of the SVs and associated control circuitry for the loss of vacuum and turbine trip features. The licensee discussed the adequacy of the testing requirements with the work order planning group. The licensee is currently reviewing the best method to stroke CV-0733.

The inspectors discussed the negative trend in post maintenance testing with the licensee. The PMTs appeared to have been written to verify the initial problem was repaired, **not** that the component continued to meet its design function following maintenance. In response to NRC questions, regarding the PMT program, the licensee is currently reviewing the PMT process.

III. Engineering

E1 Conduct of Engineering

E1.1 Review of Part-Length Control Rod Transient Analyses

a. Inspection Scope (37551)

The inspectors reviewed the applicability to Palisades of a generic NRC concern with part-length (P-L) control rods. Specifically, the concern involved the fuel vendor's elimination of two transient analysis events from the fuel cycle-by-cycle analysis normally performed for the Combustion Engineering (CE) plants that have the core protection calculator (CPC) digital protective systems.

To assess the applicability of this concern to the licensee and to verify any necessary corrective actions the inspectors held discussions with reactor engineering and operations department personnel. In addition, the inspectors also reviewed licensee plant procedures, Technical Specifications (TS), the Final Safety Analysis Report and operator training guides.

b. Observations and Findings

The issue involved not addressing certain control rod misoperation events. The two accident analysis of interest involved P-L control rod deviations while in the control deadband during startup and the slip of a P-L control rod from 50 percent inserted to 90 percent inserted. NRC review of this issue concluded that a single P-L control rod deviation within the deadband and the P-L control rod slip are an anticipated operational occurrence (AOO). An AOO is an event in which plant conditions may be present for this event once in the life of the plant. Therefore, it must be evaluated each cycle under all conditions allowed by TS.

The inspectors found several distinctions in plant configuration and administrative controls that make the two events in question highly unlikely at Palisades.

(1) Palisades has 20 shutdown, 21 regulating and 4 P-L control rods. The P-L control rod drive mechanisms (CRDMs), unlike the other CRDMs at Palisades, has a short drive shaft in place of a clutch. The CRDM motor and brake cannot be uncoupled from the control rod without disassembly. As a result, P-L control rods cannot drop into the core on a reactor trip, unlike other CE plants with the digital protective systems.

(2) At Palisades, the rods in question are not used for flux shaping during power operations. Licensee TS require P-L control rods to be completely withdrawn from the core (except for control rod exercises and physics tests). Administratively, P-L control rods are not exercised. Also, a P-L control rod

is considered inoperable if it is not fully withdrawn from the core and cannot be moved by its operator. By TS, if more than one control rod or P-L control rod becomes misaligned or inoperable, the reactor shall be placed in the hot shutdown condition within 12 hours. The licensee placed these restrictions on P-L control rods since it had been previously demonstrated on other CE plants that design power distribution envelopes could, under some circumstances, be violated by using P-L control rods.

The inspectors discussed with reactor engineering potential operator error scenarios. An operator could move P-L rods, but procedures do not allow it, except during startup prior to criticality. The operator would have to commit two errors to move a P-L rod. The operator would have to move the group selector switch for P-L rods, which gives an alarm, then move the joystick that would move the rod. This would also give an alarm. These actions would also be contrary to operator training.

The inspectors reviewed licensee surveillances pertaining to control rod movements. The procedures did not allow movement of P-L control rods.

c. Conclusions

The inspectors determined that the licensee's administrative and design features that pertained to part length (P-L) control rods provided sufficient control such that a reactor power excursion due to a stuck or mispositioned P-L control rod would be highly unlikely. Also, the licensee's fuel vendor had reviewed and determined that a P-L control rod event was bounded by a dropped or ejected control rod scenario in the current fuel cycle analysis report.

IV. Plant Support

R1 Radiological Protection

R1.1 Maintenance Activities and Daily Radiological Work Practices

a. Inspection Scope (71750 and 83750)

The inspectors observed radiological worker activities during various maintenance activities detailed in this inspection report, and also monitored radiological practices during daily plant tours.

b. Observations and Findings

The inspectors' observation of jobs in progress during the maintenance activities detailed above revealed that radiation protection technicians were visible at the job sites. The technicians took appropriate actions and surveys in accordance with good ALARA practices.

c. Conclusions

The inspectors concluded that radiological practices observed during the maintenance activities and plant daily walkdowns were adequate. The inspectors had no concerns. Specific observations are detailed below.

R1.2 ALARA Planning of "B" Radwaste Evaporator Maintenance

a. Inspection Scope (71750 and 83750)

The inspectors observed maintenance activities for the opening, inspecting and cleaning of the B radwaste evaporator.

b. Observations and Findings

The inspectors attended meetings and held discussion with ALARA planning, system engineering and other personnel involved with the B evaporator maintenance task. The inspectors' main focus was on ALARA practices for cleaning evaporator internals. A post maintenance critique meeting was also observed.

The inspectors noted that this job had potential for significant dose accumulation and that good ALARA planning and interdepartmental communication would be required to achieve a low total dose. Maintenance on the evaporator was being performed because of the overall poor material condition of the evaporator system. Auxiliary components were also scheduled for maintenance, besides cleaning evaporator internals. In addition, operations viewed system performance and reliability as poor.

Engineering had determined that cleaning evaporator internals would improve system performance. The method the licensee chose to clean the evaporator was hydrolazing. Based on past experience, the inspectors questioned why a citric acid flush of the evaporator was not considered. The licensee responded that although a citric acid flush was considered, it had not been fully evaluated. The assumption was environmental engineering would disapprove a citric acid flush because of the amount of mixed radwaste generated.

After accessing the evaporator internals the licensee began to hydrolaze. Due to the construction of the evaporator, the licensee found that most of the internals were not accessible for hydrolazing. At the outset of the work, the inspectors asked system engineering for work specific drawings. System engineering responded that no drawings were available and the specific vendor was thought to no longer exist. The inspectors, through discussions with the licensee, found cognizant individuals within the licensee's organization who were knowledgeable of the vendor and confirmed the inspector's supposition that the vendor still existed.

A contractor engineer had discussed how many evaporator steam tubes could be plugged. This information was not relayed rad protection to the engineers or ALARA coordinators planning the job. Although detailed drawings of the evaporator were not available, vendor personnel responsible for the original installation relayed important and previously misunderstood operational information that would have made a detailed inspection of the evaporator internals unnecessary prior to the planned maintenance.

The inspectors noted in the post maintenance critique a failure to fully investigate the feasibility of a citric acid cleaning, especially after the problems encountered with hydrolazing. The total dose expended for the evaporator internals inspection and cleaning was approximately 350 mrem. The maintenance window to allow for the reconditioning of the entire B evaporator system was large - at least three weeks. Even after the licensee knew the vendor still existed, the job continued as originally planned without considering the information supplied by the vendor.

c. Conclusions

The inspectors determined that the post maintenance critique did not fully address other options available to reduce dose during evaporator cleaning activities. Critique meeting participants characterized the evaporator cleaning as a low dose job (less than or equal to 10 mrem) when in fact the licensee had expended approximately 350 mrem for a job that may not have been required. The inspectors concluded that the evaporator cleaning job did not have proper emphasis placed on ALARA planning.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on July 7, 1997. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

R. A. Fenech, Senior Vice President,
Nuclear, Fossil, and Hydro Operations
T. J. Palmisano, Site Vice President - Palisades
G. B. Szczotka, Manager, Nuclear Performance Assessment Department
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J. P. Pomeranski, Manager, Maintenance
D. G. Malone, Shift Operations Supervisor
M. P. Banks, Manager, Chemical & Radiation Services
K. M. Haas, Manager, Training

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 83750:	Occupational Radiation Exposure

ITEM OPENED

50-255/97008-01	VIO	Exceeding licensed thermal power limits

ITEMS CLOSED

None

LIST OF ACRONYMS USED

ALARA	As Low As Reasonably Achievable
AOO	Anticipated Operational Occurrence
AVE	Average
CCW	Component Cooling Water
CE	Combustion Engineering
CFR	Code of Federal Regulations
CPC	Core Protection Calculator
CRDM	Control Rod Drive Mechanism
CV	Control Valve
DBA	Design Basis Accident
DRP	Division of Reactor Projects
EA	Enforcement Action
ECCS	Emergency Core Cooling System
FSAR	Final Safety Analysis Report
GL	Generic Letter
GPM	Gallons per minute
GOP	General Operating Procedure
HPSI	High Pressure Safety Injection
HXH	Heat Exchanger
I&C	Instrumentation & Control
LIC	Level Instrument Controller
MREM	Milli-Rem
Mwt	Megawatts Thermal
NRC	Nuclear Regulatory Commission
NRR	Nuclear Regulatory Research
NSO	Nuclear Shift Operator
PCS	Primary Coolant System
PCV	Pressure Control Valve
PDR	Public Document Room

P-L	Part-Length
PMT	Post Maintenance Test
PPAC	Periodic & Predetermined Activity Control
RV	Relief Valve
SDC	Shutdown Cooling
SIS	Safety Injection System
SV	Solenoid Valve
TI	Temperature Indicator
TIA	Task Interface Agreement
T-ref	Temperature - Reference
TR	Temperature Recorder
TS	Technical Specification
TYT	Temperature Transmitter
UFM	Ultrasonic Flow Measurement
VIO	Violation