U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-255/87022(DRP)

Docket No. 50-255 License No. DPR-20

Licensee: Consumers Power Company
212 West Michigan Avenue
Jackson, MI 49201

Facility Name: Palisades Nuclear Generating Plant

Inspection At: Palisades Site, Covert, Michigan

Inspection Conducted: August 4, 1987 through September 10, 1987

Inspectors: E. R. Swanson

C. D. Anderson

R. C. Kazmar

B L Burgess

Approved By: B. L. Burgess, Chief
Reactor Projects Section 2A

9/21/87
Date

Inspection Summary

Inspection on August 4, 1987 through September 10, 1987 (Report No. 50-255/87022(DRP))

Areas Inspected: Routine, unannounced inspection by resident inspectors of followup of previous inspection findings; operational safety; maintenance; surveillance; physical security; radiological protection; reportable events; regional requests and special inspections; Generic Letters and Information Notices.

Results: Of the areas inspected no violations or deviations were identified. An unresolved item was identified regarding the acceptability of the licensee's low temperature overpressure protection system and the submittal of a justification for continued operation was suggested. An additional unresolved item was identified for indeterminate diesel generator operability due to voltage response time, pending further testing. Open items were identified relating to control of safety related work and submittal of a certain Technical Specification change request.

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DETAILS

1. Persons Contacted

Consumers Power Company (CPCo)

F. W. Buckman, Vice President, Nuclear Operations
#*D. P. Hoffman, General Manager
K. W. Berry, Director, Nuclear Licensing
J. G. Lewis, Technical Director
R. J. Nicholson, Executive Manager, PE&C
W. E. Garrity, Engineering Manager, PE&C
M. R. Wade, Configuration Control Project Manager, PE&C
*R. D. Orosz, Engineering and Maintenance Manager
R. M. Rice, Operations Manager
*D. W. Joos, Administrative and Planning Manager
*C. S. Kozup, Licensing Engineer
D. J. Malone, Licensing Analyst
*R. E. McCaleb, Quality Assurance Director
*K. M. Haas, Reactor Engineering Supervisor
*R. A. Fenech, Operations Superintendent
*R. A. Vincent, Director, Plant Safety Engineering
*T. J. Palmisano, Plant Engineering Supervisor

U. S. Nuclear Regulatory Commission, Region III

C. E. Norelius, Director, Division of Reactor Projects
M. J. Virgilio, Director, Directorate III-1, NRR
T. V. Wambach, Project Manager, NRR
W. G. Guldemond, Chief, Reactor Projects Branch 2
B. L. Burgess, Chief, Reactor Projects Section 2A
#*E. R. Swanson, Senior Resident Inspector, Palisades
*C. D. Anderson, Resident Inspector, Palisades
Z. Falevits, Reactor Inspector
R. C. Kazmar, Project Inspector

*Denotes those present at the Management Interview on September 14, 1987.

#Denotes those present in Region III for the Management Meeting on August 25, 1987.

Other members of the Plant staff, and several members of the Contract Security Force, were also contacted briefly.

2. Followup on Previous Inspection Findings:

(Closed) Open Item 255/85013-09(DRP): Revisions to System Operating Procedure SOP 22, "Emergency Diesel Generators". The licensee has conferred with the vendor during their evaluation of the first concern raised by the inspector pertaining to prelube pump failure and have revised the procedure which now includes a note in Section 7.3, "Engine Lube Oil System", stating that "Failure of the prelube pump does not render the diesel inoperable". The statement "Engine lube oil prelube pump failure will be cause to run the diesel every eight hours to provide lubrication", previously contained in Section 4.0.h, has been eliminated.

In regard to the inspector's second concern, that the procedure does not provide adequate guidance relating to an overheating event, Section 5.0.d of the revised procedure instructs the operator to remove engine load and run the engine unloaded for three minutes prior to shutting it down following an overheating event. This is in agreement with the vendor operating manual.

The final concern of the inspector that related to inadequate procedural guidance for maintenance of proper oil level in the governor resulted in a procedural revision (Section 7.5.1.a) that incorporates the recommendations of the vendor.

(Closed) Open Item 255/86014-06(DRP): Discrepancy between FSAR and ONP-21 regarding the prescribed time for initation of shutdown cooling. An analysis completed by the licensee on June 11, 1986, concludes that the discrepancy between FSAR Subsection 14.15.4 and ONP-21 (currently EOP 8.0) as to when the plant can go on shutdown cooling (SDC) is actually an ambiguity in the FSAR Subsection 14.15.4.

Palisades FSAR Section 14.15, Steam Generator Tube Rupture (SGTR), assumes the plant trips at approximately 15 minutes into the incident and at 30 minutes the reactor operator starts a 75 degrees per hour cooldown, taking about 3.1 hours for the primary coolant system (PCS) to reach 325 degrees F. Subsection 14.15.4, STGR with loss of offsite power (LOOP), conservatively assumes both that the radiological release will not be terminated until the SDC system is placed into operation (3.6 hours) and that there will be continuous steaming to the atmosphere through both SGs. According to the licensee's analysis, their assumptions are ultra-conservative in that the subsection also states, "Isolation at 30 minutes is considered the most probable case...." The same subsection of the FSAR also states that, "... even for the extreme case in which isolation is not assumed before cooldown of the reactor coolant system to 325 degrees F (3.6 hours), the potential radiation dose absorbed offsite is below limits set by the guidelines of 10 CFR 100..." The licensee's analysis also states that the FSAR requirement to cooldown in 3.1 hours (75/hr) at which time the release is terminated when the plant is placed on SDC is an extreme case assumed in the FSAR and could have been predicated on postulating failure of the affected SGs main stream isolation valve. The analysis points out that Standard Review Plan (SRP), Subsection 15.6.3, does not require the postulation of a worst single component failure in addition to LOOP, and the SGTR offsite dose for Palisades is not sensitive to when the plant is placed on SDC if the affected SG is isolated and the operator follows guidance given in the operating procedures for plant cooldowns.

In addition, the current Emergency Operating Procedure (EOP) for a SGTR is 5.0 which superseded EOP 8.2. ONP-21, itself has been replaced with EOP 8.0, "Loss of Forced Circulation". Neither of the EOPs reviewed by the inspector require a cooldown rate of 75 degrees F, as assumed in FSAR Section 14.15; however, EOP 8.0 states in a note that if avoidance of reactor upper head voiding is preferred, a rapid cooldown (75 degrees F to 85 degrees F) should be used. According to the licensee's analysis, EOP 8.0 reflects guidance given in Combustion Engineering (CE) emergency procedure guidelines (CEN-199), "Effects of Vessel Head Voiding During Transients and Accidents in CE NSSS's" which states, "...voids in the upper head region, although not desirable, are not an operational concern...." This guideline also concludes that any potential void formation during plant transients addressed is not great enough to impair reactor coolant circulation or core coolability.

The adoption of EOPs 5.0 and 8.0 and information provided in the analysis, resolves the discrepancy originally noted between ONP.21 and FSAR Subsection 14.15.4.

(Closed) Unresolved Item 255/86031-03(DRP): Temporary Change Notices (TCN) to procedures were being used frequently and possibly inappropriately. An example of an inappropriate TCN use was cited as violation 255/86035-136(DRP). This Unresolved Item is being closed since the TCN issues and the corrective actions will be reviewed during closure of the violation.

No violations or deviations were identified.

3. Operational Safety

a. The inspectors observed control room activities, discussed these activities with plant operators, and reviewed various logs and other operations records throughout the inspection. Control room indicators and alarms, log sheets, turnover sheets, and equipment status boards were routinely checked against operating requirements. Pump and valve controls were verified to be proper for applicable plant conditions. On several occasions, the inspectors observed shift turnover activities and shift briefing meetings.

   Tours were conducted in the turbine and auxiliary buildings, and central alarm station to observe work activities and testing in progress and to observe plant equipment condition, cleanliness, fire safety, health physics and security measures, and adherence to procedural and regulatory requirements. A portion of the inspection activities were conducted at times other than the normal work week.

   An ongoing review of all licensee corrective action program items at the Event Report level was performed.

b. On August 10, 1987, at about 8:00 a.m. the Shift Supervisor (SS) authorized work on the 1-1 Diesel Generator (DG) lube oil pressure switch (MO EPS - 24704596) to repair a leaking fitting. While touring the plant at about 9:00 a.m., the inspector observed the ongoing work which raised a concern since the SS did not declare the DG inoperable and it appeared that the work could prevent the auto-starting of the DG. While the inspector completed the tour, a discussion between the system engineer and the SS resulted in the awareness that a mistake was made, i.e. that a Service Water pump (P-7C) on the opposite train was concurrently inoperable for maintenance and the DG was in fact inoperable contrary to Technical Specification 3.7.2. At 9:50 a.m. the Service Water pump was returned to service, and at 10:42 a.m. the 1-1 DG was test started to verify operability. In the condition which existed between 8:00 and 9:50 a.m., the plant was operating under action requirement 3.7.2 of the Technical Specifications which requires a shutdown within 12 hours to hot shutdown. Although the licensee unknowingly entered this action requirement, it was satisfied by returning the Service Water pump to service. The plant then remained under the Limiting Condition for Operation of 3.7.2, which allows a DG to be inoperable for seven days per month. The DG was restored to service on August 10, 1987, at 3:10 p.m. after evaluating the quality of the repair work performed. This event is an example of poor review/controls in authorizing work. Preliminarily the cause of this event appears to be a lack of attention to detail on the part of the SS and persons planning the work. The root cause is still under review, and a Licensee Event Report will be submitted.

c. On August 12, 1987, the licensee identified that work in progress on a containment isolation valve (CK-CRW 408) would result in violation of containment integrity requirements of Technical Specifications 3.6.1 and 1.4. The work, which had been released by the Shift Supervisor (SS) was not recognized as affecting containment integrity. Action was taken to restore the valve alignment to normal and test the valve. No problems were identified with the valve and the maintenance order was cancelled. Similar to the event above (subparagraph b), this event raises concerns for the adequacy of work planning and for the SS releasing work where he is not knowledgeable of the work scope and its impact on safety. Corrective action was initiated by the licensee to review the responsibility of the SS in releasing work with each SRO and to review the Operations Department work planning process (D-PAL-87-115). These actions will be tracked as an Open Item (255/87022-01(DRP)).

d. On August 14, 1987, the licensee reduced power from 100 percent to approximately 2 percent power to allow troubleshooting of the "B" primary coolant pump (PCP) motor. The motor had a minor oil leak on the upper oil reservior which necessitated a containment entry to add oil and determine the location of the leak. The power reduction was necessary to reduce radiation levels to minimize personnel exposure. Investigation of the "B" PCP motor identified oil leaking from a sight glass on the upper oil reservoir and a leak around a manway cover. The licensee repaired the leaks, refilled the reservoir and inspected the other PCPs for leakage prior to synchronizing to the grid at 6:24 a.m. on August 15, 1987.

e. While returning to full power on August 17, 1987, at 4:04 a.m., operators were alerted to an electro-hydraulic control (EHC) leak when the standby EHC pump started and the main turbine governor valves started closing. The reactor was manually tripped. Operators noted no equipment operating problems during the trip. The plant was operating at 68 percent power while performing a primary coolant system leak rate calculation. The NRC was notified of the trip at 4:57 a.m. on the same day.

   The EHC fluid leak was identified on the supply line to the #4 governor valve at a midpoint of the flexible coupling. Flexible hose was recently installed on all governor valve EHC lines as a corrective measure to eliminate cracking due to vibration that had been experienced with the hard piping. The EHC fluid leak identified on the supply line to the #4 governor valve was determined to be a result of improper installation and/or application of the flexible hose. Since time consuming engineering and procurement would be required to replace the flexible hose, the licensee decided to reinstall hard piping on the EHC lines until the planned EHC line modification can be installed during the October 1987 maintenance outage.

   The reactor was made critical at 5:37 p.m. on August 18, 1987. The generator was connected to the grid at 4:33 a.m. on August 19, 1987, after a delay caused by generator voltage control problems. The voltage regulator would not allow the matching of the machine and bus voltages so the voltage control was switched to an alternate supply. The licensee later decided to isolate the #4 turbine governor valve to preclude another failure until additional action can be taken to resolve the vibration phenomenon. This action limits power to 94% until the October outage.

f. Following a satisfactory between the doors test of the personnel air lock, a door seal test was performed on the inner door and was declared a failure at 4:45 p.m. on August 20, 1987. A four hour non-emergency 10 CFR 50.72 report was made and corrective maintenance was performed. Both doors of the airlock open into containment so that pressure seals the doors closed. During the airlock integrity test, pressure is applied between the doors. To prevent the inner door from unseating, strongbacks are installed to hold the door closed. The seals had been compressed by the strongbacks, and after cleaning and adjustment a satisfactory seal test was performed on the inner door at 3:15 a.m. on August 21, 1987.

Additional testing was also required on the escape lock, which was used to enter containment to perform the work on the inner door seal. During preparation for the escape lock test, the inner seal of the escape airlock was found to be leaking air. This condition was identified by personnel inside the airlock when air was whistling through the inner seals. This condition was in violation of the licensee's Technical Specifications for containment integrity, and required prompt corrective action. The licensee closed the outer door, repaired the inner seals and satisfactorily retested the airlock.

On August 20, 1987, the licensee was planning to enter through the outer personnel airlock door to make repairs on the failed inner door seal of the same airlock. When the inspector was notified of their plans, they were encouraged not to willfully violate containment integrity. Plans were then made to enter through the escape airlock. The licensee plans corrective action regarding the containment airlock testing and repair methods and additional discussions on Technical Specification interpretation is planned.

g. On August 23, 1987 at 6:28 a.m., operators received a spurious generator alarm and seconds later a turbine trip on loss of load and a reactor trip. The plant was operating at a steady state of 93 percent power. Investigation determined that the "Trinistat" generator voltage regulator power amplifier had a failed potentiometer which caused a loss of the generator field. The "Trinistat" drawer was repaired, load tested for several hours, and determined to operate satisfactorily.

On August 25, 1987 at 2:22 a.m., operators took the reactor critical. The turbine was placed on line after steam generator chemistry hold points were satisified at 3:15 p.m. on the same date.

Other conditions which were noted after the trip and resolved include: the correct alarm setpoint associated with an alarm operators expected to receive was verified; the "B" main feed pump seal leakage was evaluated as acceptable; and the auxiliary feedwater pump control valve CV-0727A packing was adjusted, the stem lubricated, and stroke tested satisfactorily.

h. On August 25, 1987, at 1:15 p.m. when the reactor was critical at about 2% power, the average coolant temperature (Tave) dropped from the desired 533 degrees F to below the Technical Specification (TS) 3.1.3.(c) limit of 525 degrees F for less than a minute. This transient condition was a result of the combined effects of a rapid xenon burnout rate, boration to maintain criticality, low control rod worth due to rod height, turbine/generator voltage regulator testing and feed pump startup. Of the two similar events which preceded this, one was nearly identical in root cause, i.e., operator error. The basis for the TS requirement is for the moderator temperature coefficient effect which could act in reverse to that normally expected at the beginning of core life. Since the core is nearly at midlife, the significance of the violation is minor except in its reflection on the operator's control of the plant.

Corrective actions planned to prevent recurrence include revision of the TS to reflect an allowance of 15 minutes for this condition to exist before a shutdown is required. No citation is being issued due to the peculiar history this issue has had. As a result of prior occurrences, a TS change request was submitted on February 5, 1985. After the subsequent withdrawal request by the licensee the revision request was withdrawn by letter on May 29, 1986, wherein NRR stated that the time allowance to take action was already provided for in TS 3.0.3. As explained in Generic Letter 87-09, failure to meet the provisions of an LCO or its associated action requirement results in a violation and entry into the shutdown provisions of TS 3.0.3. Additionally, the 15 minute shutdown requirement of the Standard Technical Specifications (which the original change request was patterned after) provides improved control for operation in this unanalyzed condition. The TS change request should be reinitiated to allow proper and timely action to be taken in future events (Open Item 255/87022-02(DRP)).

i. While at 92% power, at 1:10 p.m. on September 1, 1987, an inadvertent auxiliary feedwater actuation signal (AFAS) occurred due to personnel error. The "A" and "C" AFW pumps started and ran for approximately 30 seconds before the operators secured them after determining that it was not a valid signal. During performance of the AFAS logic surveillance test (MI-39), an instrument and control technician mistakenly actuated two sensor channels as opposed to one sensor and one actuation channel. The sensor panels are in close proximity to the actuation panels. This procedure has been run monthly for more than a year with no similar problems. The 10 CFR 50.72 notification was made at approximately 2:20 p.m. The planned corrective actions include a procedure review by the Instrument and Control department to investigate if procedure enhancements can be made and an evaluation by the Human Performance Evaluation System Coordinator for possible human error prevention recommendations.

j. While conducting computerized Diesel Generator (DG) loading modeling studies, the licensee determined that when the last load is sequenced on the 1-1 DG the voltage would not recover due to the sluggish voltage regulator response seen during the last surveillance on February 26, 1986. As a result of this discovery the licensee declared the 1-1 DG inoperable at 11:00 a.m. on September 3, 1987, and entered a 7 day limiting condition for operation.

Unloaded voltage regulator response testing was conducted on September 4, 1987, and adjustment to the voltage regulator "stability" or damping characteristic was made in accordance with the vendor's recommendation. Based on initial investigation it appears that the "as found" condition existed since initial setup. The licensee plans to conduct loaded DG response testing in the "as found" condition during the next shutdown to determine whether the DG was operable for design basis accidents since initial startup. Surveillance testing methods will also be reviewed and revised accordingly. Further investigation of this event will be tracked as Unresolved Item 255/87022-03(DRP).

This event was expediently brought to management's attention and dispositioned in a conservative and forthright manner.

k. On September 3, 1987 at 4:30 p.m., a licensee Quality Assurance audit identified a chemical sampling requirement in Technical Specification (TS) Table 4.2.1 which was not being met. Prompt action was taken to obtain the sample while reviewing the circumstances. It was determined and verified by the inspector that the requirement to sample the "B" Safety Injection Tank (SIT) (T-82B) was put into the TS by Amendment 74 on January 21, 1983, with applicability limited to core cycle 5 while excessive SIT leakage was being experienced. When the expired conditions were removed by TS Amendment 101 on February 10, 1987, the sampling provision was not removed as it should have been. The NRC has agreed that an increased sampling frequency for T-82B is no longer required, and that the provision should be deleted. The licensee submitted a request for TS revision on September 11, 1987. No enforcement action by the NRC is warranted, but the late discovery of the error does have implications of inattentiveness to detail in the plant TS review process.

l. Containment pressure had been running higher than usual during early September 1987. On September 6, 1987, pressure reached 0.9 psig. Pressure did not respond when containment temperature was decreased. Upon exceeding 1.0 psig, the plant must be shut down in accordance with a proposed Technical Specification. The usual vent pathway for containment noncondensible gases is through an open clean waste receiver tank manway located inside containment, into the vent gas collection header (VGCH) and out the plant stack. It is suspected that the VGCH is plugged. Maintenance planning is in progress. An alternate pathway has been established off the VGCH upstream of the blockage, through the chemistry lab drain tank into an open drain upon which a suction is drawn by the auxiliary building ventilation system which discharges to the plant stack. This pathway has made the laundry room an airborne contamination area since the tank and drain are located there. Pressure has decreased through use of this alternate pathway. Investigation of the problem is continuing and modifications are planned to improve the efficiency of the VGCH.

m. At 8:45 a.m. on September 10, 1987, an inadvertent "B" Low Pressure Safety Injection (LPSI) pump start occurred during a preventive maintenance activity. A plant electrician was cleaning the Design Basis Accident sequencer contacts when his screwdriver slipped, bumping the "B" LPSI pump contact, starting the pump. The Control Operator stopped the pump within approximately one minute. The maintenance activity was stopped while evaluating methods to prevent recurrence. The 10 CFR 50.72 notification was made at 9:14 a.m. on the same date. Corrective action includes the planned replacement of the sequencers with a type not requiring preventive maintenance and a review of the activity to determine if there are any procedural improvements possible.

No violations or deviations were identified.

4. Maintenance

The inspectors reviewed and/or observed the following selected work activities and verified whether appropriate procedures were in effect controlling removal from and return to service, hold points, verification testing, fire prevention/protection, radiological controls, and cleanliness where applicable:

a. Lube Oil Pressure Switch replacement on 1-1 DG (EPS-24704596).
b. Starting Air Pressure Gage replacement on 1-1 DG (EPS-74701979).
c. Preventive Maintenance on P-41 Fire Pump (FPS-24704376).
d. Corrective Maintenance & Calibration of (CV 0701) Main Feedwater Regulating Valve (FWS-24705153).

No violations or deviations were identified.

5. Surveillance

The inspectors reviewed surveillance activities to ascertain compliance with scheduling requirements and to verify compliance with requirements relating to procedures, removal from and return to service, personnel qualifications, and documentation. The following test activities were inspected:

a. MO-22 Inservice Test Procedure - High Pressure Safety Injection Pumps.
b. DWO-1 Daily Control Room Surveillance.
c. SHO-1 Operators Shift Surveillance.

No violations or deviations were identified.

6. Physical Security

The inspectors observed physical security activities at various locations through out the protected and vital areas including the Central and

Secondary Alarm Stations. Periodic observations of access control activities including proper personnel identification, badging and searches of personnel, packages and vehicles were conducted. The inspectors verified appropriate security force staffing and operability of search equipment. Protected and vital area boundaries were toured to verify maintenance of integrity. Illumination was verified to be adequate to support patrol and Closed Circuit Television (CCTV) monitor observations. CCTV monitor clarity and resolution were also observed. The inspectors periodically verified that appropriate compensatory measures were taken for degraded or inoperable equipment and breached boundaries.

No violations or deviations were identified.

7. Radiological Protection

The inspectors made observations and had discussions concerning radiological safety practices in the radiation controlled areas including: verification of radiation levels and proper posting; accuracy and currentness of area status sheets; adequacy of and compliance with selected Radiation Work Permits and high radiation procedures; and the ALARA (As Low AS is Reasonably Achievable) program. Implementation of dosimetry requirements, proper personnel survey (frisking) and contamination control (step-off-pad) practices were observed. Health Physics logs and dose records were routinely reviewed.

No violations or deviations were identified.

8. Licensee Event Reports

Through direct observations, discussions with licensee personnel, and review of records, the inspectors examined the following reportable events to determine whether: reportability requirements were met; immediate corrective action was accomplished as appropriate; and corrective action to prevent recurrence has been accomplished.

(Open) LER 255/84022: In September 1984, the licensee discovered that the containment building air temperature was routinely greater than the 104 degrees F value assumed in the Main Steam Line Break (MSLB) analysis. The LER references a subsequent analysis that determined that the 55 psig design pressure limit would not be exceeded unless the average initial temperature was in excess of 137 degrees F. On July 4, 1983, a temperature of 138 degrees F was measured in the containment dome. Since the LER was issued on November 21, 1984, other analyses have been done which indicate that a higher temperature may be acceptable. An LER revision is necessary to address the final analysis and associated temperature limit.

A Technical Specification (TS) change request had not been submitted to include containment temperature but one is being considered again.

System Operating Procedure SOP-24, "Ventilation and Air Conditioning System", requires a plant shutdown in accordance with TS 3.0.3 if the temperature exceeds 150 degrees F, though currently there is no TS limit or action statement. Operators record containment temperatures each shift on one log sheet and daily on another, but of those operators interviewed, none were aware of the shutdown requirement of SOP 24. The daily log sheet does reference the temperature limit and shutdown requirement. There is no alarm for high containment temperature, therefore no associated Alarm Response Procedure exists to direct the operators to shutdown the plant for high temperatures. These issues and inconsistencies must be resolved prior to closure of this LER.

(Closed) LER 255/85023, Revision 1: Low temperature overpressure protection (LTOP) system setpoint error. As documented in Paragraph 9 in this report, several discoveries were made relating to the inadequate protection provided by the existing LTOP system as a result of reduced pressure/temperature operation curves. The LER requires updating to include a basis for continued operability and outline corrective actions being taken to resolve this issue. These actions will be tracked under Unresolved Item 255/87022-04(DRP).

(Closed) LER 255/87017: Two compensatory flow estimates were not made as required by Technical Specification 3.24.2 during a waste gas batch release while flow indicator FI-1121 was inoperable. Poor communications between the Radiological Material Control Supervisor and the Operations crew resulted in the failure to make the estimates at four hour intervals following the initial estimate. The activity contained in the waste gas release was a fraction of 10 CFR 20, Appendix B limits. Health Physics Procedure HP 6.6, "Evaluation and Release of Waste Gas Decay Tanks", has been revised by temporary change to note on the Release Authorization that if FI-1121 is inoperable, the primary side auxiliary operator must make the flow estimates. The document is supplied to the Shift Supervisor for each batch. System Operating Procedure SOP 18-A, "Radioactive Waste System - Gaseous", has been temporarily changed adding a note referring to the above. As allowed by 10 CFR 2, Appendix C, no notice of violation will be issued for this violation because it: was licensee identified, fits a Severity Level IV or V classification, was reported, corrective actions have been taken and the violation could not have been reasonably expected to have been prevented by previous corrective actions.

(Closed) LER 255/87018: Following a rapid power reduction due to an electro-hydraulic control system leak, the reactor was critical at approximately ten percent power in violation of Technical Specification (TS) 3.1.3.c. During the power reduction, the "A" main feedwater regulating valve failed to close and when the turbine was tripped, the moisture separator reheater control valves failed to close. Additional information can be found in Inspection Report 255/86015(DRP). These two failures resulted in overcooling the primary coolant system temperature below the TS minimum of 525 degrees F for approximately thirty seconds.

The equipment problems have been corrected. As allowed by 10 CFR 2, Appendix C, this TS violation is not being cited due to the equipment failures, identification by the licensee, fitting Severity Level IV or V, and being reported, corrected and not expected to have been prevented by previous corrective actions.

(Closed) LER 255/87021: The reactor was manually tripped due to an oil leak from the upper reservoir of primary coolant pump P-50D. The event is described in Inspection Report 255/85018(DRP) Paragraph 3.c. A nearly circumferential crack was found at the root of a thread, flush with the discharge head of the back stop lube oil pump. The cracked discharge line was replaced and the similar lines for the other primary coolant pumps were inspected with no abnormalities noted. A failure analysis of the crack is ongoing.

(Closed) LER 255/87022, Revision 1: During plant startup while the reactor was critical, primary coolant temperature decreased below the Technical Specification (TS) 3.1.3.c minimum of 525 degrees F for approximately fifteen seconds. The overcooling was the result of overfeeding the "A" steam generator (SG) due to main feedwater regulating valve CV-0701 being open when the "B" main feewater (FW) pump was started. Operators noticed the SG level rise and closed CV-0701. The usual position for CV-0701 for this plant condition (plant startup) is closed though there is no procedural guidance. The Control Operator was not aware of CV-0701 being in the open position prior to starting the FW pump. The "Turbine Generator Startup From Hot Standby" procedure, GOP 4, has been changed to include a verification that the main feedwater regulating valves are closed and their associated controllers are in manual. No violation is being issued for the reasons outlined above in Paragraph 3.h.

(Closed) LER 255/87024: A startup transformer failed, resulting in the loss of offsite power and a manual reactor trip. This event and report were reviewed by the Augmented Inspection Team, which documented their inspection activities in report 255/87019(DRP). The event report was well written and discussed all pertinent aspects of the event and corrective actions.

No additional violations or deviations were identified.

9. Regional Requests and Special Inspections

(Closed) Temporary Instruction (TI) 2500/19: Inspection of licensee's actions taken to implement Unresolved Safety Issue A-26 (Reactor Vessel Overpressure Protection). A review was conducted of the design of the Low Temperature/Overpressure Protection (LTOP) system, which was implemented under Facility Change FC-404. The analysis performed by Energy Incorporated for Palisades in June 1977, was reviewed and it was noted that the design criteria were conservative but would require reanalysis within two to ten years. Drawings were reviewed to verify that single failure criteria were implemented in the design and that the system is not susceptible to event initiators which could also render the LTOP system inoperable. The analysis results indicated that one Power Operated Relief Valve (PORV) with a set relief pressure of 400 psig had sufficient capacity to provide overpressure protection from the design basis accident (primary coolant pump (PCP)) start with a hot steam generator.

In 1985 the licensee discovered that the 1980 and 1985 revisions to the Primary Coolant System's (PCS) pressure/temperature (P/T) operating limit curves had not been factored into the setpoint of the LTOP system. The setpoint was determined to be non-conservative based on the curve shift resulting from reactor vessel irradiation. A review of the 1977 analysis showed that protection from exceeding the 10 CFR 50, Appendix G limit (P/T curves) was not provided for inadvertent High Pressure Safety Injection (HPSI) pump start or PCP start with a hot steam generator (70 degrees F above PCS temperature). Administrative controls of pulling HPSI pump fuses during plant cooldown and racking out and tagging PCP breakers were implemented at this time. The above discovery and corrective actions were documented in Licensee Event Report 255/85023 on November 18, 1985.

A subsequent update of the LER was made on December 23, 1986, after it was discovered on August 29, 1986, that a previously acceptable overpressure scenario (charging/letdown imbalance) was now not acceptable. An inadvertent start of all three charging pumps on a Safety Injection Signal (SIS) was found to exceed the existing P/T curve. As compensatory measures, administrative controls were implemented to limit the number of operable charging pumps to two. This final corrective action dealt with the last of the possible transient initiators that were known at that time.

Several weaknesses were found in the licensee's existing LTOP system and controls. During the inspector's review it was noted that the administrative controls which were implemented are not single-failure proof. An overpressurization event could result from a personnel error in the implementation of the controls and an initiating event. This does not meet the design criteria established for the LTOP system. These administrative controls, relied upon for over two years, have not yet been submitted to the NRC for inclusion in the Facility license. The inspector also postulated an inadvertent SIS which starts the two operable charging pumps with the single failure being the Shutdown Cooling relief valve. The expected system response, similar to that experienced in 1981 and documented in LER 255/82004, would exceed the existing P/T curve for a brief period. The PORV setpoint was verified to be 375 psia, but no setpoint derivation was able to be located by the end of the inspection. Although a design criteria, post modification testing did not demonstrate the operability of each train upon the loss of AC and/or DC power to one or both trains, nor verify the actual ability of the valves to operate at low pressures. Since installation, several events have demonstrated the ability of the valves to operate and a weakness of the indication system. Loss of power to the PORVs and control circuits would not be annunciated and not likely detected for some time. Corrective action was taken to have the power supplies verified on a shiftly basis while LTOP is required.

Review of administrative procedure SOP-1 determined that precautions exist concerning solid-water operation, minimizing the temperature differential between the steam generators and the PCS (maximum of 21 degrees allowed), and disabling the High Pressure Safety Injection pumps from inadvertently starting. Annunciators exist to warn operators of both the overpressure condition and the operation of the LTOP system. Surveillance procedures (MO-27 and RM-56) were also reviewed and determined to be adequate and appropriately scheduled.

In summary, the Palisades plant is protected from low temperature/over pressure transients largely by administrative controls, several of which have not been incorporated into the TS, and by the LTOP system, which because of reduced reactor vessel ductility, no longer protects the vessel from all potential transients as required by 10 CFR 50, Appendix G. Resolution of this issue is two fold: A justification for continued operation should be submitted (in update of the LER 255/85023 if desired), and plans for future analysis to resolve the unanalyzed condition should be outlined. NRR will have the lead for evaluation of this submittal. This is an Unresolved Item (255/87022-04(DRP)).

(Closed) TI 2515/84: Primary coolant system pressure isolation (Event V) valves order verification. By review of the licensee's Technical Specification (TS) it was verified that the license modification was entered as required by the order dated April 20, 1981, and further modified by Amendment No. 72 on December 21, 1982. Testing of these twelve valves was reviewed in Inspection Reports 255/84020(DRS) and 255/86010(DRP) where it was concluded that licensee surveillance procedures SO-9 "PCS Pressure Isolation Check Valves" and GOP-13 "Primary Coolant System Leakage Calculation" provide adequate testing to comply with TS 4.3.h. Records of the last test (November 29, 1986) for the subject valves were reviewed and found to meet TS requirements. The as-found leakage was recorded and no test anomalies were noted that would indicate improper or inadequate testing.

Acceptance criteria were established for operability of the check valves. The surveillance procedure provide assurance that leakage of one half the TS limit or a step increase in leakage will result in corrective maintenance.

(Closed) TI 2515/86: Natural Circulation Cooldown implementation (Generic Letter 81-21). The licensee's implementation of committments made in their response to Generic Letter 81-21 dated November 16, 1981, were verified by review of Emergency Operating Procedure (EOP 8.0) "Loss of Forced Circulation Recovery" which has superseded Off-Normal Procedure (ONP-21). The adequacy of the procedure as well the extent of operator knowledge in using the procedure was demonstrated by the successful handling of the recent July 14, 1987, event involving loss of offsite power. A discrepancy between the FSAR analysis for steam generator tube rupture coincident with loss of offsite power and ONP-21 (Open Item 255/86014-06(DRP) regarding the prescribed time for initiation of shutdown cooling during natural circulation is discussed in Paragraph 2.

10. Generic Letter Followup

(Closed) Generic Letter 87-06: "Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves". The licensee's response dated June 5, 1987, provided, as requested, a list of pressure isolation valves, along with a description of the measures performed to assure the valve integrity as an independent reactor coolant pressure boundary barrier, and the acceptance criteria used for the valves. The letter was routed to applicable plant management staff.

11. Information Notices

The inspector reviewed selected NRC Information Notices (INs) and the licensee's mechanism for determination of applicablity, distribution and corrective actions. Since the last review, documented in Inspection Report 255/86031(DRP), the licensee has drafted a revision to the "Nuclear Operating Experience Review Program" procedure, NADP XIX-3, that governs the handling of NRC Information Notices. Although implemented, final approval of the revision is in progress. Concerns raised in the referenced inspection report are addressed in the revised procedure by an increased delineation of the responsibilities of the Nuclear Operating Experience Review Coordinator (NOERC). These include the maintenance of a status tracking system, expediting the dissemination of notices deserving immediate attention to appropriate plant personnel, and assignment of a Lead Evaluator (LE) for each document screened by Plant Safety Engineering (PSE) as applicable. Initial screening of notices for potential impact on the safety and reliability of plant operations continues to be the responsibility of the PSE Section Head and the NOERC.

The revision also provides for the forwarding of notices to the Training Review Tracking Committee, of which the PSE Section Head is a member.

Notices which are screened as applicable are assigned a LE and the normal completion date for the evaluation is within sixty days of initial screening. Longer or shorter completion dates may be assigned depending on the urgency of the item. In the event the LE is confronted with a particular point that is beyond his technical knowledge or experience, it is his responsibility to contact individuals or organizations having the required information.

It is anticipated that adherence to the procedural revision will assure proper distribution, timely assignment of INs for action and proper evaluation by knowledgeable persons.

Licensee action regarding the following INs were reviewed by the inspector and were found to be adequate and timely. A brief synopsis of the licensee's engineering analysis is also provided.

a. IN 87-01: RHR Valve Misalignment Causes Degradation of ECCS in PWRs (Issued January 6, 1987). Palisades shutdown cooling configuration was determined to be substantially different from the subject Westinghouse plants and precludes isolation of two injection lines in a manner similar to that addressed in the notice.

b. IN 87-04: Diesel Generator Fails Test Because Of Degraded Fuel (Issued January 16, 1987). Diesel generator fuel oil filters are replaced every six months. A Technical Specification surveillance provides for monthly sampling and tests are conducted for viscosity and percent of moisture and sediment. In addition, the rate of fuel oil turnover is high being replaced about every two to four weeks. Consideration is being given to using a biological additive.

c. IN 87-08: Degraded Motor Leads In Limitorque DC Motor Operators (Issued February 4, 1987). Palisades does not have any Limitorque DC motor operators.

d. IN 87-10: Potential For Water Hammer During Restart Of Residual Heat Removal Pumps (Issued February 11, 1987). Engineering analysis determined that due to the shutdown cooling design configuration, the system is not susceptible to the event as described in the notice.

e. IN 87-12: Potential Problems With Metal Clad Circuit Breakers, General Electric Type ARF-2-25 (Issued February 13, 1987). This breaker type is found only in the feedwater purity system and a preventive maintenance test requires monthly operation.

f. IN 87-14: Actuation Of Fire Suppression System Causing Inoperability Of Safety-Related Ventilation Equipment (Issued March 23, 1987). The configuration of the ventilation equipment fire protection at Palisades requires the manual valving in of fire water. An operator must connect flexible fire hoses and open two valves on each line. It was, therefore, concluded that a similar flooding event is very unlikely.

g. IN 87-23: Loss Of Decay Heat Removal During Low Reactor Coolant Level Operation (Issued May 27, 1987). Analysis by PSE concluded that practices and methods in use by the licensee were generally sufficient to prevent a loss of shutdown cooling similar to the type described in the IN. The analysis also addressed in considerable detail the six NRC recommendations in the notice. Further detailed review of this issue will be conducted following the licensee's response to the related 10 CFR 50.54(f) request (Generic Letter 87-12).

12. Management Meeting

On August 25, 1987, in the Region III offices, representatives of Consumers Power Co., led by Dr. F. W. Buckman, presented a status of various action items including the development of the Configuration Management Program to Mr. C. E. Norelius and certain NRR and Region III staff. The licensee presented the overall Configuration Management Program scope, schedule and overview. Details were given for the work scopes of the electrical system upgrade and plant design basis efforts. The licensee also presented a summary and overview of the screening and tracking system for the 10 CFR 50.54(f) letter commitments. Items that require clarification or that they wish to take exception to, were discussed along with the updating of the System Functional Evaluation program. Additional communication and correspondence related to these later two topics is continuing.

13. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, violations or deviations. Unresolved Items disclosed during the inspection are discussed in Paragraphs 3.j and 9.

14. Open Items

Open items are matters which have been discussed with the licensee, which will be reviewed further by the inspectors, and which involve some action on the part of the NRC or licensee or both. Open items disclosed during the inspection are discussed in Paragraphs 3.c and h.

15. Management Interview

A management interview was conducted on September 14, 1987, following the conclusion of the inspection. The scope and findings of the inspection were discussed. The inspector also discussed the likely information content of the inspection report with regard to documents or processes reviewed by the inspectors during the inspection. The licensee did not identify any such documents/processes as proprietary.