U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-255/90015(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: June 1 through July 16, 1990

Inspectors: E. R. Swanson
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Reactor Projects Section 2A

8/9/90
Date

Inspection Summary

Inspection on June 1 through July 16, 1990 (Report No. 50-255/90015(DRP))
Areas Inspected: Routine unannounced inspection by the resident inspectors of: actions on previously identified items; plant operations; reactor trips; maintenance; surveillance; reportable events; design changes; baseline inspection of replacement steam generators; public meeting; and, NRC Regional requests. No Safety Issues Management System (SIMS) items were reviewed.
Results: Of the ten areas inspected, no deviations were identified. One licensee identified violation(s) concerning a missed surveillance test was identified during LER review (Paragraph 7.c). Two unresolved items concerning piping stress analysis activities were identified during review of an LER (Paragraph 7.c). Three open items were identified concerning a design change and modification (Paragraph 8).

The inspection disclosed weaknesses in the licensee's pursuit of the reactor system vent valve leakage problem, concerns with the PCS drain line stress analyses, and less than adequate closure of certain Configuration Control Project (CCP) issues.

The inspection noted strengths in the licensee's work coordination, planning and waiver of compliance associated with the pressurizer heater transformer; identification of potential containment structural stresses resulting from thermal loads; and the CCP identification of minor problems in plant design, construction or configuration.

DETAILS

1. Persons Contacted

Consumers Power Company

+D. W. Joos, Vice President, Energy Supply
+G. B. Slade, Plant General Manager
+R. M. Rice, Plant Operations Manager
*D. J. VandeWalle, Technical Director
*R. D. Orosz, Engineering and Maintenance Manager
*K. M. Haas, Radiological Services Manager
J. L. Hanson, Operations Superintendent
R. B. Kasper, Mechanical Maintenance Superintendent
K. E. Osborne, System Engineering Superintendent
R. M. Brzezinski, I&C Engineering and Maintenance Superintendent
*C. S. Kozup, Technical Engineer
J. R. Brunet, Licensing Analyst
D. J. Malone, Plant Projects Supervisor
*W. L. Roberts, Senior Licensing Analyst
K. A. Toner, Plant Projects Superintendent
+M. A. Savage, Public Affairs Director

Nuclear Regulatory Commission (NRC)

+H. Brent Clayton, Chief, Reactor Projects Branch 2
*+E. R. Swanson, Senior Resident Inspector
*J. K. Heller, Resident Inspector
+R. Lickus, Chief, State & Government Affairs
+R. J. Marabito, Region III Public Affairs Officer
+R. C. Pierson, NRR Technical Assistant
*+M. L. Dapas, NRR Operations Engineer
J. A. Gavula, Region III Inspection Specialist
J. F. Schapker, Region III Inspection Specialist

+Indicates some of those attending the July 11 licensee sponsored Public Meeting to discuss the Steam Generator Replacement Project

*Denotes some of those present at the Exit Interview on July 20, 1990

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

2. Actions on Previously Identified Items (92701, 92702)

(Closed) Open Item 255/89034-01: A Garlock 204 expansion joint located in the 1-1 Diesel Generator service water line appeared to be improperly elongated during installation. Deviation Report 89-169 documents that: the expansion joint was improperly elongated during installation, the alignment was not correct, and the control rods were not installed. In addition, similar problems were found for Diesel Generator 1-2. Work Orders 24004262 and 24004265 were written to resolve the problems.

No violations, deviations, unresolved or open items were identified.

3. Operational Safety Verification (71707, 71710, 42700)

Routine facility operating activities were observed as conducted in the plant and from the main control room. Plant startup, steady power operation, plant shutdown, and system(s) lineup and operation were observed as applicable.

The performance of Reactor Operators and Senior Reactor Operators, Shift Engineers, and Auxiliary Equipment Operators was observed and evaluated. Included in the review was procedure use and adherence, records and logs, communications, shift/duty turnover, and the degree of professionalism of control room activities.

Evaluation, corrective action, and response for off normal conditions or events, if any, were examined. This included compliance to any reporting requirements.

Observations of the control room monitors, indicators, and recorders were made to verify the operability of emergency systems, radiation monitoring systems and nuclear reactor protection systems, as applicable. Reviews of surveillance, equipment condition, and tagout logs were conducted. Proper return to service of selected components was verified.

a. General

The unit began the reporting period in power operations and remained at 80 percent power except for a brief period as described in Paragraph 3.b below.

b. Pressurizer Heater Transformer Failure & Shutdown

On June 8, 1990 at 12:55 a.m., pressurizer heater transformer No. 15 failed due to an internal ground. Transformer No. 15 powers one half of the pressurizer heaters and was powered from bus 1E. Technical Specification 3.1.1.i requires that: a minimum of 375 KW of pressurizer heater capacity be available from bus 1D and 1E; the capacity be restored in 72 hours or the plant be in hot shutdown in 12 hours; and, that the primary coolant system shall not be maintained greater than 325 degrees Fahrenheit without 375 KW heater capacity from bus 1D and 1E. Since an action statement was not provided for maintaining the plant above 325 degrees Fahrenheit, the licensee considered that Technical Specification 3.0.3 applied once the plant was in hot shutdown.

The licensee declared an Unusual Event at the time of the event due to the anticipated shutdown. The shutdown was started at 10:00 p.m. on June 8 and hot shutdown was reached at 8:55 p.m. on June 9, 1990.

For reasons documented in a June 10, 1990, letter from K. W. Berry (Director of Consumers Power Nuclear Licensing) to the U.S. Nuclear Regulatory Commission, the licensee requested a seven day waiver of compliance to maintain the plant in hot shutdown while a replacement pressurizer heater power supply was obtained.

The Unusual Event was terminated at 12:45 p.m. on June 11, 1990, when a Waiver of Compliance was granted from Region III. The plant staff was observed to be very dedicated and thorough in their design effort and in the execution of the necessary work activities. The unit was returned to service on June 16 at 7:02 a.m., after installing a replacement pressurizer heater transformer.

c. Head/Pressurizer Vents

During this reporting period, PIA-1066 "Gaseous Vent Pressure Gauge" slowly increased to equal the primary coolant system pressure and alarmed, indicating that one of the isolation valves from the head or pressurizer vent system was leaking. PIA-1066 is located between the first and second isolation valves, from the head and pressurizer vent system. The second isolation valve appeared to be leak tight as evidenced by the constant reading on PIA-1066 and the lack of leakage indicated by the pressurizer quench tank level or the containment atmosphere temperature/humidity. The inspector reviewed Work Order (W.O.) 24003660, which was written to resolve the elevated PIA-1066 pressure during the next refueling outage. The work plan contained instructions to recalibrate PIA-1066, but did not specify testing to determine which valve was leaking.

A similar problem was identified approximately a year ago, and was documented in Inspection Report No. 50-255/89012. At that time the licensee planned to perform a special test (T-210, "Reactor Vent and Pressurizer Vent System Flush and Valve Leakage Test") during the April/May 1990 Maintenance outage to determine if one of the first isolation valves was leaking. T-210 was never performed, however, a calibration of PIA 1066 was performed per W.O. 24903529. The calibration sheet indicated that PIA-1066 was working satisfactory.

The inspector discussed W.O. 24003660 with the appropriate System Engineer Section Chief, who indicated that T-210 will be scheduled for the upcoming refueling/steam generator replacement outage. The inspector has reviewed T-210 and questions if the test can detect minor leakage since the test is performed for 10 minutes per valve with system pressure at 1/10 normal operating pressure. The records show that PIA-1066 increased to primary coolant system operating pressure over several shifts. This issue was discussed at the exit meeting.

d. Standing Order 54

While reviewing the Primary Coolant Gas Vent systems the inspector found that the system was not addressed by a Technical Specification. When the system was installed, a Technical Specification Change Request (TSCR), dated August 30, 1982 and supplemented by letter dated April 21, 1988, was submitted. While the TSCR was being evaluated, the TSCR was included in Standing Order 54. Standing Order 54 was the licensee's administrative mechanism to control systems or equipment that the licensee was committed to maintain. Standing Order 54 refers to the TSCR as "supplemental" Technical Specifications. Due to NRC interim position on restructured Technical Specifications the licensee was permitted to withdraw the TSCR. The NRC letter dated May 15, 1989, that concurred with the licensee withdrawal, stated that the licensee will maintain the system under administrative control and any changes thereto will be reviewed under 10 CFR 50.59. The cover letter of Standing Order 54 (Revision 26 - dated November 15, 1988) stated that a supplemental Technical Specification may be waived provided approval was obtained from both PRC & the General Plant Manager. The cover letter also stated that such waivers were expected to be rare. The inspector discussed the waiver process with the site licensing department and several managers, who could not state that a waiver would be reviewed under 10 CFR 50.59. This was discussed at the exit since the hidden requirement to perform a 10 CFR 50.59 review could be a trap to future plant management. At the exit interview the licensee indicated that an engineer was assigned to look into the matter.

Paragraph 15 of Standing Order 54 lists three surveillance requirements for the Primary Coolant Gas Vent System. The inspector reviewed these requirements and verified that the licensee has implemented these requirements. However, one surveillance requirement requires a verification of flow through the reactor coolant vent system vent paths during cold shutdown. This is accomplished by RO-112 "Reactor Head/Pressurizer Vent Flow Check". The licensee has chosen to indirectly verify flow by cycling the valves in a sequence that pressurizes and depressurizes the header pressure gauge. The basis document for RO-112 stated that since the Technical Specification does not specify the quantity of flow, the test was written to verify the paths are capable of passing fluid. The inspector questioned the licensee to readdress the indirect flow method and assure that it meets the intent of RO-112.

e. Operability of Primary Coolant System Leakage Detection Systems

During testing of the containment gas radiation sample solenoid valve (SV-1822) after replacement of the containment penetration connector, it was found that the circuit fuse block was removed. Power to this solenoid valve is required to open the valve to obtain containment gas radiation readings. Apparently the licensee had taken numerous readings and grab samples through this line, not knowing that the valve was not opening when the switch was positioned. Numerous small solenoid valves in the plant have no positive indication of position, however, some indicate whether voltage is applied to open or close the valve. This condition apparently existed for over two and a half years. The circuit had been walked down in October 1987 and the missing fuse block had been noted as removed. Inadequate review was done by the Configuration Control Project, which resulted in the absent fuse block condition being incorporated into the revised drawings. As this is a licensee identified issue, the licensee is taking corrective action to investigate and resolve this failure. Under the Systematic Evaluation Program, the licensee's commitment to submit Technical Specifications (TS) for primary system leakage detection inside containment was documented in the IPSAR (NUREG-0820), Item 4.15.2. The NRC did not approve their request due to the lack of accuracy of the existing instrumentation and the apparent need to install additional instrumentation. Further NRC action permitted withdrawal of the TS Change Request (NRC letter dated May 5, 1989) based on a commitment to conduct further review and include the topic in the Restructured TS submittal due in early 1990, now expected in mid-1991.

f. <u>Primary Coolant System Cold Leg Drains</u>

During a Region III review of Primary Coolant System cold leg drain modifications completed during a previous outage, a Region III inspector identified that an incorrect input assumption was used for a cold leg drain line stress calculation. This error was the subject of conference calls between NRC (Region III and NRR) and Consumers Power (plant and corporate), during the week of June 11.

Using the correct assumption the licensee confirmed that the safety related portion of the cold leg drain piping met the FSAR stress limits. To assure that additional errors were not made, a third party review of the calculations was performed. The third party review did not identify any additional problems. See Paragraph 7 for additional discussion on this topic.

g. <u>Containment Temperature Exceeds Structural Design Values</u>

During the review of a Safety Evaluation, the licensee identified that actual containment ambient and FSAR (Chapter 14) accident analysis temperatures exceed the FSAR (Chapter 5) structural design values for containment. The structural analysis was based on an operating temperature of 104 degrees Fahrenheit (versus 135 degrees Fahrenheit allowed by current Technical Specifications), and the DBA temperature of 283 degrees Fahrenheit (versus MSLB temperature calculated at over 400 degrees Fahrenheit). A statement in the FSAR Chapter 5 indicates 60 percent of the total stresses on the equipment hatch are thermal. In recent years the licensee had modified Chapter 14 analysis as a result of identifying that actual ambient containment temperature approached 130 degrees Fahrenheit, but had not considered the need to look at the structural analysis. A containment analysis performed by Bechtel, which incorporates the appropriate temperatures, is currently under review. This action was underway to support the steam generator replacement containment opening. A "quick look" engineering judgement indicated no immediate problems or operability concerns. The licensee reviews continue and will be documented in an LER if the reporting threshold is met.

h. Control Room Cooling

As a result of reviews performed by the CCP of the Control Room (CR) HVAC, it was identified that inappropriate heat load assumptions had been the basis for a series of analyses that had reduced service water flows to the chillers. Heat loads such as internal; heat transmission through walls; outdoor air ventilation; fan motor heat; and filter unit electric heater loads were not considered. Initial reanalysis demonstrated that the CR temperature would be maintained below 75 degrees F with a SW Temperature of 75 degrees. A refined analysis was performed which demonstrated operability of the CR HVAC with a SW inlet temperature of 80 degrees which resulted in a final CR temperature of about 81 degrees. These values are within the operability envelope established in the Technical Specifications for the SW and CR temperatures. The licensee is planning to submit an LER on this topic, and further review will be done at that time.

i. System Walkdowns

(1) During this inspection period a detailed walkdown of the battery and associated 125 Volt DC electrical distribution system was conducted. This included a review of the licensee's recently completed Safety System Design Confirmation (SSDC) audit of the DC electrical distribution system as part of the licensee's ongoing Configuration Control Project (CCP). Review of the SSDC audit indicated that it was comprehensive, identifying a number of concerns. An apparent significant finding from this audit was the lack of adequate separation between redundant channels of DC equipment in the Cable Spreading Room. This issue had previously been addressed in NRC Inspection Report No. 50-255/90010 which reviewed the licensee's implementation of their CCP for selected systems. During the course of that inspection the licensee stated that a previous commitment to submit a voluntary Licensee Event Report (LER), documenting and evaluating potentially significant cable separation discrepancies, would instead be documented by letter to NRC Region III. Several inconsistencies with the values for various battery parameters specified in the Design Basis Document (DBD), surveillance procedures, and vendor manual were also identified in the SSDC audit. The licensee has initiated corrective action for many of the audit findings and was currently evaluating the safety significance of those findings requiring more extensive corrective action.

(2) The inspector verified operability of accessible portions of the High Pressure Safety Injection System, Low Pressure Safety Injection System, and Containment Spray System by verifying system alignment using applicable portions of Revision 19 to Checklist 3.8, "Engineered Safeguards System Checklist", and Revision 33 to Checklist 3.9, "Engineered Safeguards Administrative Control Verification". No items were found that degraded the system.

No violations, deviations, unresolved or open items were identified.

4. <u>Reactor Trips</u> (93702)

During the pressurizer transformer outage startup, an unplanned loss-of-load reactor trip occurred on June 15 at 8:10 p.m. The trip, which occurred with all rods fully inserted, was caused because a checklist did not reflect a recent off-site power modification. Apparently, the modification removed a loss-of-load bypass switch that the checklist intended to be in-service. The checklist established plant conditions to allow a manual activation of the turbine loss-of-load circuit when the automatic actuation occurred. All systems responded as designed. The checklist was completed after establishing the proper system configuration.

The inspector has no additional questions but will review this subject when the LER is issued.

No violations, deviations, unresolved or open items were identified.

5. <u>Maintenance</u> (62703, 42700)

Maintenance activities in the plant were routinely inspected, including both corrective maintenance (repairs) and preventive maintenance.

Mechanical, electrical, and instrument and control group maintenance activities were included as available.

The focus of the inspection was to assure the maintenance activities reviewed were conducted in accordance with approved procedures, regulatory guides and industry codes or standards and in conformance with Technical Specifications. The following items were considered during this review: the Limiting Conditions for Operation were met while components or systems were removed from service; approvals were obtained prior to initiating the work; activities were accomplished using approved procedures; and post maintenance testing was performed as applicable.

The inspector reviewed the electrical maintenance backlog and found it to be well managed. Reactive issues were dealt with promptly. Good cooperation and coordination was exhibited between plant and contract electricians during the repowering of the pressurizer heaters.

The following activities were inspected:

a. Replacement of SV-1452 in the fuel oil supply line to the 1-2 Diesel Generator Day Tank (Work Order (W.O.) 24002570, Functional Equivalency Evaluation).

b. Repair of MV-EV102 on the B Waste Evaporator (W.O. 24004140).

c. Planning for resolution of elevated reading on PIA-1066 per Work Order 24003660. See Paragraph 3.c for discussion of this Work Order.

d. Planning for replacement of Garlock 204 expansion joints per Work Orders 24004262 and 24004265.

No violations, deviations, unresolved or open items were identified.

6. <u>Surveillance</u> (61726, 42700)

The inspector reviewed Technical Specifications required surveillance testing as described below and verified that testing was performed in accordance with adequate procedures, that test instrumentation was calibrated, that Limiting Conditions for Operation were met, that removal and restoration of the affected components were properly accomplished, that test results conformed with Technical Specifications and procedure requirements and were reviewed by personnel other than the individual directing the test, and that deficiencies identified during the testing were properly reviewed and resolved by appropriate management personnel.

The following activities were inspected:

a. MI-02 Reactor Protective Trip Units.

b. QO-20A Inservice test of Low Pressure Safety Injection System.

c. DW-013 Local Leak Rate Test for inner/outer personnel air lock seals.

d. DWO-1 Daily Control Room Surveillance.

e. SHO-1 Operators Shift Surveillance.

The following completed surveillances were reviewed:

f. ME-12 Station Battery checks dated 4/2/90, 5/3/90 and 5/30/90.

g. QE-9 Quarterly Diesel Fire Pump Battery checks dated 3/22/90 and 5/15/90.

The inspector reviewed various supporting documentation for QE-9. Palisades Technical Specification 4.17.2.2.b states that operability of the fire pump diesel engines and starting 24 volt battery banks and charger shall be demonstrated by verifying at least once per three months that the specific gravity of the starting battery bank is "appropriate for continued service". The basis document for QE-9 stated that the battery manufacturer has defined a specific gravity of 1.200 as being "appropriate for continued service". The basis document further stated that the low limit for acceptable test results shall be defined as 1.220 based on the manufacturer's recommended value for initiating recharge. Additionally, test results shall be considered unacceptable but the battery still operable if the specific gravities are between 1.200 and 1.220. Finally, for specific gravities less than 1.200, "battery operability determination shall require further investigation and evaluation". The acceptance criteria for QE-9 stated, "A battery is considered inoperable and shall be replaced if the arithmetic difference between the highest and lowest corrected specific gravities is equal to or greater than 0.050 points." The acceptance criteria does not address the case where the battery specific gravity is less than 1.200. In view of the above discussion, the battery could have a specific gravity of less than 1.200 and not be declared inoperable. This was discussed with the system engineer and at the exit interview. Review of the two most recently conducted diesel fire pump battery surveillances did not identify any specific gravities below 1.200.

h. FE-5A/5B Battery Performance Test dated 8/8/88.

i. RE-48 18 Month Diesel Fire Pump Battery check dated 3/14/89.

No violations, deviations, unresolved or open items were identified.

7. Reportable Events (92700, 92720)

The inspector reviewed the following Licensee Event Reports (LERs) by means of direct observation, discussions with licensee personnel, and review of records. The review addressed compliance to reporting requirements and, as applicable, that immediate corrective action and appropriate action to prevent recurrence had been accomplished.

a. (Closed) LER 255/90006: Calculated Cold Leg Drain Thermal Stresses Result In Pipe Stresses In Excess of FSAR Allowable Due to Design Error. As a result of as-built walkdowns, in preparation for an upcoming modification, the licensee found discrepancies in the configuration of the reactor coolant system cold leg drain lines. Subsequent analyses determined that the piping exceeded FSAR allowables in non-safety related portions of the system. The safety related portions of the system were not isolated from the safety related portions by an anchor and, as such, the effect of the over-stressed piping had to be evaluated. In addition, supports on the system were apparently over-stressed and had other non-safety related piping attached to them which complicated the evaluation. Because of recent problems with CPCo's design control and verification process, the NRC requested that the licensee submit the analysis which demonstrated that the safety related portions of the piping met FSAR stress limits. A Region III Regional review of

EA-SP-33315-PS-01, Revision 0, dated May 21, 1990, determined that the nozzles for cold leg drain line 1A and 1B were modeled differently. The stress intensification factor (SIF) for nozzle 1A was 2.0, whereas, the SIF at 1B was 2.1. Also, the second element for the 1A nozzle was modeled as 2.375 inch diameter with a 0.343 inch wall thickness, whereas, the second element for the 1B nozzle was modeled as 4.00 inch diameter with a 1.00 inch wall thickness. Both nozzles should have had the same modeling. Aside from these modeling discrepancies, an erroneous assumption was made by the analyst regarding thermal displacements of the nozzles. Because of an apparent lack of understanding of the reactor coolant system (RCS) thermal expansion design input calculation, the analyst assumed that the lateral expansion of the 1A and 1B nozzles relative to each other was negligible. By making this assumption, approximately 0.8 inches of lateral displacement was neglected in the analysis. The NRC considered this a significant oversight in light of the recently cited inadequacies in the licensee's design control and verification process.

The licensee revised and submitted EA-SP-33315-PS-01. The NRC found that the revised calculation addressed the discrepancies and the piping appeared to meet the design criteria. However, the anchor bolts for supports HC1-H4A and HC1-H5 had exceeded the allowable interaction equation by approximately five percent. A detailed review determined that a conservative linear function was used for these evaluations and by using an acceptable non-linear function, all anchor bolts appeared to be acceptable.

For the above issue, the following items are considered unresolved:

(1) The initial problem must be evaluated. Initial statements by the licensee indicated that a 1981 modification of support HC1-H5 caused the piping overstress. Although this may have contributed to the problem, the NRC inspector could not rule out the potential that there were deficiencies in the original design and installation of the RCS drain lines (Unresolved Item 255/90015-01(DRS)).

(2) Commitments made by the licensee, as a result of recent design control enforcement actions, indicated that a third party review would be conducted on piping analyses until the controlling design specification was revised. If this work was not done in accordance with the revised specification, the licensee may not have adequately implemented their commitment since a third party review was not initially performed. However, if this work was done in accordance with the revised specification, the adequacy of the corrective actions taken to address the recent design control deficiencies may not be sufficient (Unresolved Item 255/90015-02(DRS)).

Pending a review of licensee actions to address the above issues, these are considered Unresolved Items.

b. (Closed) LER 255/90009: Automatic actuation of Auxiliary Feedwater system resulted from spurious operation of the atmospheric dump valves while in hot standby. This event was discussed in Inspection Report No. 50-255/90014, Paragraph 3.a. The LER was found to accurately describe the event and the corrective actions taken by the licensee. Additionally, the temporary modifications made will be removed during the upcoming refueling outage when the Reactor Regulating Pressurizer level and pressure control, and steam dump control systems will be replaced with programmable digital controllers (FC-861).

c. (Closed) LER 255/90010: Technical Specification required sampling of the primary coolant system for iodine activity was delayed due to personnel error. This event was reviewed in Inspection Report No. 50-255/90014, Paragraph 3.e. In accordance with 10 CFR 2 Appendix C.V.G, a Notice of Violation will not be issued for the violation of the Technical Specification since: it was licensee identified; classified as Severity Level IV or V; reported if required; not a willful violation; and will be corrected, including measures to prevent recurrence, in a reasonable period of time (Closed Violation (NV6) 255/90015-03(DRP)).

One licensee identified violation, no deviations or open items, and two unresolved items were identified.

8. Design Changes (37700)

The inspector reviewed Facility Change 906, "Modify Main Feedwater Regulatory Valves (CV-0701 and CV-0703) and Main Feedwater Bypass Valves (CV-0734 and CV-0735) control circuit to close on Containment High Pressure Signal," to ascertain that this design change and modification was done in conformance with the requirement of the Technical Specifications, 10 CFR 50.59 and the licensee Quality Assurance Program. The design change was implemented because the licensee determined that the analysis for a main steam line break transient described in their FSAR was not limiting for containment pressure. In their original analysis they had analyzed for a guillotine-type steam line break as the bounding case for peak containment pressure. Feedwater isolation currently occurs on a low steam generator pressure. A followup analysis was performed on a small break which would allow a slower blowdown of the faulted steam generator. This resulted in a delayed isolation of the feedwater which would allow more steam to be released into containment for a longer period of time, resulting in a higher peak containment pressure than previously analyzed. The following were discussed with the licensee:

a. The safety evaluation review determined that a Technical Specification Change was not required. This conclusion was made because the Technical Specification did not address the main feedwater regulation or bypass valves. The inspector questions if this conclusion was correct since this modification and a modification made approximately 5 years ago to close the valves on low steam generator pressure were implemented to mitigate the consequences of a main steam line break (small or large) on containment peak pressure. The licensee was asked to review this item and determine if a Technical Specification Change request was required. This is an Open Item pending completion of the license review. (Open Item 255/90015-04(DRP)).

b. The following comments pertain to post modification test T-FC-906-001, "Test Containment High Pressure Trip Activation for Feedwater Valves CV-0701, CV-0735, CV-0703, and CV-0734:

(1) Steps 5.1.2 & 5.2.2 required that two normally open contacts be blocked by placing shrink tubing over the contacts. The procedure provided steps to remove the shrink tubing but did not require verification (either visually or by test) that the contacts were still functional. The inspector asked if post removal verification was required to detect any damage to the contacts during installation or removal of the shrink tubing. This is an open item pending completion of the licensee review (open item 255/90015-05(DRP)).

(2) The test did not require stroke timing of the valves. After the test was completed the valves were stroked and the time recorded. A note was attached stating that the times were done cold, however, the stroke time was well below the time assumed in the accident analysis for valve closure. The inspector could not find an analysis documenting that the cold stroke will equate to the time assumed in the accident analysis during accident operating temperature and pressure. The inspector asked if timing while the plant was cold would assure that the valve would function within assumed time during an accident. This is an open item pending completion of the licensee review (open item 255/90015-06(DRP)).

(3) The test verified that the valve will close on containment high pressure and verified that the closure on steam generator low pressure was not affected. However, the licensee chose to jumper out the remaining outputs from the containment high pressure relay and not verify that the modification would not affect the ones previously in use.

c. The design package contains copies of procedures that require revision because of this modification. One procedure that required revision was RO-12 "Containment High Pressure System Test". This test verifies that components activate when a containment high pressure signal is introduced. The proposed change requires verification that the feedwater valve goes shut but does not require verification of the stroke time. The operability of these valves is determined by verification of closure and stroke time to assure that the assumptions in the accident analysis are met. At the exit interview the licensee stated that stroke timing is part of the section XI testing. The inspector asked if the section XI testing verified the time from the initial input to final closure of the valve.

9. Public Meeting

On July 11, 1990, Consumers Power Company sponsored an informational public meeting at the local high school concerning the steam generator replacement project. The meeting was moderated by the local township supervisor, a presentation was made by David Joos, Vice President Energy Supply, and both he and Eric Swanson, Senior NRC Resident Inspector, answered questions from the audience. The meeting was attended by about 80 people and their questions and concerns were answered in a professional manner. NRR, Regional management and public affairs representatives were in the audience. The tone of the meeting was generally friendly.

10. Regional Requests (92701)

a. During this inspection the inspector reviewed the licensee plans to detect and combat Zebra Mussels. The presence of Zebra Mussels in the Great Lakes has been confirmed. They have caused water intake problems at both power and non-power facilities located on the lakes. The licensee has outlined the following actions:

Completed actions:

(1) A Betz Bio Box was installed in the service water inlet stream to detect the veliger larva stage of the zebra mussel. Samples are collected monthly and sent to a laboratory for analysis.

(2) Concrete blocks are positioned in the service water inlet pipe and in the dilution water pipe going to the mixing basin. They are periodically inspected to detect the adult stage of the Zebra Mussel.

To date, the licensee has not detected any Zebra Mussels, but they have been identified at the D.C. Cook plant located approximately 30 miles to the south.

Future Corrective Actions:

(3) An application package will be prepared and submitted to the Michigan Department of Natural Resources for approval to use a molluscicide (approved by the U.S. Environmental Protection Agency), which is currently in use at one of the licensee's coal burning electrical generating facilities.

(4) Design and installation of the appropriate equipment to apply the molluscicide or other treatment chemicals will be completed as necessary.

b. Check Valve Program Inspection (73756)

At the request of the NRC Regional Office a review of the licensee's check valve program was conducted. The scope of the licensee's check valve program was found to encompass the Pressure Isolation Valves and others deemed important to safety (Q-listed). Prioritization of all "Q" safety related check valves is complete. Prioritization of other valves (important to safety) will be completed in 1991. The inspection phase has begun but will not be completed for six years. A sample of valves were selected and reviewed for adequacy of testing/disassembly and inspection. Deficiencies were identified in the completeness of the inspection records, indicating less than adequate reviews were being performed. The licensee is evaluating, but has not implemented any of the non-intrusive testing methods.

Preventive and predictive maintenance has been generally effective in identifying excessively worn components and focusing future maintenance efforts. Deficiencies identified during inspection and testing do not always result in the initiation of a corrective action document in addition to the corrective maintenance work order. The licensee considers their planned trending program and reviews to be sufficiently comprehensive to identify generic problems and bring significant issues to the attention of management within the framework of the check valve program. The licensee's procedure, "Equipment Performance Monitoring and Trending" (EM-20), has not been modified to address the parameters to be trended for check valves.

In summary, a plan has been developed to implement a comprehensive, predictive and preventive maintenance program including appropriate trending and prioritization but implementation is a long term project. Consumers Power Company Presidential level management involvement in the issue has been evident through correspondence and periodic briefings.

No violations, deviations, unresolved or open items were identified.

11. Baseline Inspection of the Replacement Steam Generators

a. Background

The licensee will replace steam generators during the next outage (October 1990). In preparation for the steam generator replacement, baseline eddy current, ultrasonic and liquid penetrant examinations and other modifications of the replacement steam generators were performed. Eddy current examination was being performed by the licensee assisted with contracted services from Allen Nuclear Associates (ANA). Ultrasonic and liquid penetrant examinations were performed by the licensee's nondestructive examination personnel.

b. Inspection

The NRC inspector observed the eddy current and ultrasonic examinations. Eddy current examination was performed using Zetec MIZ 18 data acquisition with bobbin, 8x1, and motorized rotating pancake coils (MRPC). A 100 percent examination of the steam generator tubes was performed with the bobbin coil, which was supplemented with a random sample of 8x1 coil examinations. The MRPC was utilized to evaluate any distorted indications detected with bobbin or 8x1, which could not be analyzed with the bobbin or 8x1 generated data. In addition, visual inspections were performed to evaluate indications which were accessible.

The licensee identified distorted indications with the bobbin coil located approximately four to six inches from the tube sheet face. The licensee performed visual examinations and confirmed the ET indications were due to weld spatter adhered to the ID of the tubes. The visual examination confirmed the presence of metallic nodules adhered to the ID of the tubing. The ET signal response was indicative of the visually observed ID distortion.

Other eddy current data observed during the examination were identification of four tubes which had not been expanded during fabrication, and possible permeability indications in various locations within the tubes of the steam generators. The licensee contacted the vendor who dispositioned the unexpanded tubes as acceptable for service. The permeability indications were examined with a magnetically biased probe and found to be acceptable. Further examination of one of these indications was made using the MRPC probe which confirmed the bobbin indications were caused by permeability (magnetic properties of tubing material disrupting the eddy current signal) and were not flaws.

The NRC inspector also observed the installation of the tube sheet templates, cleanliness control, ET data collection, management, and ET analyses.

The Ultrasonic examination (UT) of the replacement steam generators' vessel welds was observed by the NRC inspector. Observations included the calibration of UT equipment, setup, and examinations in progress. Certification of the UT inspectors was reviewed and found to be in compliance with NDT-A-02 and SNT-TC-1A requirements. The ultrasonic equipment was verified to be in calibration. Verification of screen height and amplitude control linearity was made, and beam spread measurements were completed in accordance with procedural and Code requirements. Calibration blocks were fabricated in accordance with ASME Code requirements. The UT inspectors performed the examinations with diligence and proper technique. Indications detected were sized in accordance with Code requirements. No Code rejectable indications were detected. Recordable indications were documented in accordance with applicable procedure requirements.

Inspection data will be maintained to establish baseline data for Inservice Inspections (ISI) required by ASME Section XI. The NRC inspector also observed vendor radiographs of recorded UT indications to assist in interpretation of the cause of flaws detected. Reader sheets for the radiographs were not available; however, position markers on the radiographs were matched to vessel markings to locate the areas of interest. Review of the radiographs where reportable UT indications were found disclosed possible slag inclusion or porosity as the reflecting mechanism. The slag inclusions and porosity were within Code allowable dimensions.

The NRC inspector noted that the radiographs were not adequately protected to assure longevity. (No papers separating the film, film envelopes damaged and previous handling practices caused film damage.) The licensee representative indicated retrieval of the reader sheets from the vendor has been requested and measures to protect the film will be made.

c. Procedures Review

The NRC inspector reviewed the following procedures:

Procedure No.	Title	Revision
NDT-G-06	Installation of Tube Sheet Templates - Palisades Steam Generators	1
NDT-ET-17	Eddy Current Examination - Motorized Rotating Pancake Coil (MRPC)	0
NDT-ET-18	Eddy Current Examination - MIZ-18 Pull Through Techniques	0
EM-09-11	Palisades Eddy Current Procedure for Data Management for the Replacement Steam Generators	0
ECT-SG-1	Inservice Inspection Steam Generator Plan Inspection	18
SQAP-041	Palisades Eddy Current Examination for the Replacement Steam Generators	0

Procedure No.	Title	Revision
NDT-PP-26	Preservice Examination Program Plan	0
NDT-UT-11	Ultrasonic Examination of Steam Generator Vessel Welds	1
NDT-UT-01	Ultrasonic Examination of Ferretic and Austenitic Piping and Branch Connection Welds	8
NDT-UT-12	Ultrasonic Examination of Nozzle to Vessel Welds and Nozzle Inner Radius Sections	1
NDT-PT-01	Liquid Penetrant Examination	8

The licensee's preservice examination of the replacement steam generators complies with ASME Code and regulatory requirements. The licensee inspectors demonstrated professionalism in their performance of the required nondestructive examinations.

No violations, deviations, unresolved or open items were identified.

12. Open Items

Open Items are matters which have been discussed with the licensee, which will be reviewed further by the inspector, and which involve some action on the part of the NRC or licensee or both. Open Items disclosed during the inspection are discussed in Paragraph 8.

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 Paragraph 7.a.(1) and (2).

12. Management Interview (30703)

The inspector met with licensee representatives (denoted in Paragraph 1) on July 17, 1990, to discuss the scope and findings of the inspection. In addition, the inspector also discussed the likely informational content of the inspection report with regard to documents or processes reviewed by the inspector during the inspection. The licensee did not identify any such documents/processes as proprietary.