September 18, 2001

Mr. Douglas E. Cooper
Site Vice President
Palisades Nuclear Plant
Nuclear Management Company, LLC
27780 Blue Star Memorial Highway
Covert, MI 49043-9530

SUBJECT: PALISADES NUCLEAR GENERATING PLANT
NRC SPECIAL INSPECTION REPORT 50-255/01-11

Dear Mr. Cooper:

On August 9, 2001, the NRC completed part of a Special Inspection at your Palisades Nuclear Generating Plant regarding an active steam/primary coolant system leak from a through wall crack in the upper housing assembly for Control Rod Drive Mechanism 21. The enclosed report documents the inspection findings which were discussed on August 9, 2001, with you and other members of your staff.

This inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Beginning on about June 9, 2001, your staff detected and monitored an increase in primary coolant system leakage until the plant was shut down from full power to Hot Standby (Mode 3) on June 20, 2001. On June 21, 2001, the resident inspector and a member of your staff identified the active steam/primary coolant system leak from Control Rod Drive Mechanism 21. Your staff subsequently placed the plant in Cold Shutdown (Mode 5) and assembled a project team to evaluate the occurrence, assess extent of condition, determine root cause, and develop strategies for restoration. Non-destructive examination revealed that an axially-orientated flaw had propagated from the inside surface of the eccentric reducer to pipe weld on the housing of Control Rod Drive Mechanism 21. On June 27 and July 1, 2001, during examinations for extent of condition, your staff identified indications of cracks on two additional control rod drive mechanism housings. Your staff believed the cause to be transgranular stress corrosion cracking on all the housings.

Based on criteria specified in Management Directive 8.3, "NRC Incident Investigation Program," and Inspection Procedure 71153, "Event Followup," a Special Inspection was initiated in accordance with Inspection Procedure 93812, "Special Inspection." The purpose of the special inspection was to assess your plant staff's performance, and to the extent practicable, independently validate your staff's efforts in areas including root cause determination, adequacy of repair, and corrective actions. A charter was developed to focus the inspection effort on determining: (1) Sequence of Events; (2) Root Cause; (3) Safety Significance; (4) Extent of

Condition; (5) Adequacy of Repair Methodology; (6) Adequacy of Overall Corrective Actions; (7) Similarity with Other Leakage Issues; (8) Quality of Non-destructive Testing; and (9) Adequacy of Radiological Controls.

Because your staff was still in the process of investigating root cause, extent of condition, repair methodology, corrective actions, and similarity with other leakage issues, we could not complete the Special Inspection. Therefore, no findings of significance were identified. We will complete the Special Inspection after you have completed your activities in these areas.

There were several instances where the NRC team identified engineering issues that may not have been adequately addressed by your staff which were categorized as unresolved items.

- This discovery of stress corrosion cracking raises a question regarding the adequacy of actions to evaluate the extent of condition and corrective actions from prior cracking in the control rod drive seal housings discovered during the recently completed refueling outage.

- The design basis loading for the housings did not appear well understood by your staff in that the bending moments on the control rod drive housings may not have been adequately determined for use in the calculation of critical crack size.

- Your staff may not have adequately considered the effects that primary coolant leakage up through the control rod drive nozzle would have on the function of the control rod.

- Your staff's evaluation of a postulated control rod ejection did not evaluate the impact on the operability of adjacent rods due to forces transmitted through the interconnecting seismic supports.

- Your staff's operability evaluation of the reactor vessel missile shield being installed differently than the original design did not identify structural weaknesses that could have allowed the missile shield to fall onto the reactor vessel head during a design basis earthquake.

Based on the number of issues, the scope, depth, and rigor of your staff's engineering work appeared inconsistent. Evaluation of these issues will be conducted during this ongoing special inspection.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available **electronically** for public inspection in the NRC Public Document Room **or** from the *Publicly Available Records (PARS) component of NRC's document system (ADAMS)*. *ADAMS is accessible from* the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

Original signed by
John A. Grobe

John A. Grobe, Director
Division of Reactor Safety

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

Enclosure: Inspection Report 50-255/01-11(DRP)

cc w/encl: R. Fenech, Senior Vice President, Nuclear
Fossil and Hydro Operations
N. Haskell, Director, Licensing and Performance Assessment
R. Anderson, Chief Nuclear Officer, NMC
A. Udrys, Esquire, Consumers Energy Company
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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-255
License No:	DPR-20
Report No:	50-255/01-11(DRP)
Licensee:	Nuclear Management Company, LLC (NMC)
Facility:	Palisades Nuclear Generating Plant
Location:	27780 Blue Star Memorial Highway Covert, MI 49043-9530
Dates:	July 2 through August 9, 2001
Inspectors:	David Passehl, Project Engineer and Inspection Lead Melvin Holmberg, Mechanical Engineering Branch James Gavula, Mechanical Engineering Branch David Nelson, Plant Support Branch Sonia Burgess, Senior Reactor Analyst
Approved by:	Anton Vegel, Chief Branch 6 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000255-01-11(DRP), on 07/02-08/09/2001(DRP); Nuclear Management Company, LLC (NMC), Palisades Nuclear Plant. Special Inspection.

This special inspection was initiated to evaluate the facts, circumstances and licensee actions surrounding discovery of a through wall crack in the control rod drive housing for Control Rod Drive (CRD) 21. This was accomplished by direct observation, review of records, and discussions with personnel. The inspection was conducted by a Regional Project Engineer, two Regional Mechanical Engineering Inspectors, a Regional Plant Support Inspector, and a Regional Senior Reactor Analyst. Because this inspection was incomplete, the inspection identified no findings. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

The licensee was still in the process of investigating root cause, extent of condition, repair methodology, corrective actions, and similarity with other leakage issues. Therefore, this Special Inspection could not be completed. NRC will complete the Special Inspection after the licensee has completed their investigation. However, based on the number of substantive unresolved engineering issues identified related to root cause, extent of condition, adequacy of corrective actions/extent-of-condition reviews, and engineering evaluations/calculations, the consistency of the scope, depth and rigor of the licensee's engineering work was questioned. This discovery of stress corrosion cracking raises a question regarding the adequacy of actions to evaluate the extent of condition and corrective actions from prior cracking in the control rod drive seal housings discovered during the recently completed refueling outage. The design basis loading for the CRD housings did not appear well understood by the licensee. The forces resulting in bending moments on the control rod drive housings may not have been adequately determined for use in the calculation of critical crack size. The licensee did not adequately consider the effect that leakage flow up through the CRD nozzle would have on the function of the control rod to insert upon a reactor trip. The licensee's evaluation of a postulated control rod ejection did not evaluate the impact of that potential rod ejection on the operability of adjacent rods due to forces transmitted through the interconnecting seismic supports. The licensee discovered that the missile shield over the reactor vessel was installed differently than the original design and the operability evaluation did not identify structural weaknesses that could have allowed the missile shield to fall onto the top of the reactor vessel during a design basis earthquake.

Report Details

Summary of Plant Event

Beginning on about June 9, 2001, plant operators detected and monitored an increase in primary coolant system leakage until the plant was shutdown from full power to Hot Standby (Mode 3) on June 20, 2001. The leakage began at approximately 0.1 gpm and showed a slowly increasing trend on containment sump level instrumentation and containment gas radiation monitors. On June 21, 2001, the resident inspector and a member of the licensee's staff identified an active steam/primary coolant system leak from the eccentric reducer to pipe weld on the housing of Control Rod Drive (CRD) 21. Upon identifying the primary coolant system pressure boundary leakage, the licensee placed the plant in Cold Shutdown (Mode 5) and assembled a project team to evaluate the housing leak, assess extent of condition, determine root cause, and develop strategies for resolution. Containment sump in-leakage measured approximately 0.3 gpm just prior to the beginning of plant shutdown. Plant personnel subsequently identified additional cracks on other CRD housings.

As of August 9, 2001, plant personnel were continuing with testing and examination activities on the CRD housings. The licensee had examined 26 accessible CRD housings, out of a total of 45, using radiography, and was preparing to visually examine the interior surfaces of the remaining 19 housings using a high resolution camera. The radiographs for 26 of the housings had been reviewed, and crack indications had been identified in 23 of them, including the CRD-21 housing where the initial leak was found. Crack indications had been identified in both axial and circumferential orientation. The licensee was still reviewing and developing possible repair procedures for the housings where cracking had been found.

4. OTHER ACTIVITIES [OA]

4OA3 Event Followup (93812)

.1 Sequence of Events

a. Inspection Scope

The inspectors reviewed documentation and conducted interviews to determine the chain of events regarding the CRD housing defects.

b. Findings

b.1 Sequence of Events

On approximately June 9, 2001, operations personnel began to notice indication of elevated primary coolant system unidentified leakage. Plant personnel suspected leakage from a swagelock fitting on a sample line from the pressurizer head vent system. The fitting had been known to drip slightly while the plant was in Hot Standby (Mode 3) during startup from the Cycle 16 refueling outage that ended in May 2001. The licensee attempted to repair the fitting during startup but was unsuccessful because the leaking fitting could not be isolated with the system pressurized. On June 17, 2001, plant personnel performed a special containment entry to attempt to identify the source of the leakage. Plant personnel observed a small amount of leakage from the swagelock fitting leak but could not access the leak for measurement due to adverse radiological conditions with the plant operating at full power.

On June 20, 2001, plant operators shut down the plant from full power operation in accordance with a forced outage plan to repair the swagelock fitting leak and identify and repair the source of any other primary coolant system leaks. On June 21, 2001, plant personnel and the resident inspector identified an active steam/primary coolant system leak from a through wall crack in the housing for CRD-21 near the pipe to eccentric reducer butt-weld. As of August 9, 2001, plant personnel were continuing with testing and examination of various CRD housings toward determination of root cause, extent of condition, and corrective actions.

b.2 Leakage Detection

Plant operators performed a primary coolant system mass balance calculation daily in accordance with plant procedures to determine primary coolant system leak rate. In addition, operators routinely used the plant process computer to check volume control tank level, containment sump fill rate, and containment gas radiation level, which are indicators of primary coolant system leaks. The plant process computer also had trending capabilities. The inspectors reviewed the licensee's data and agreed that evidence of increasing primary coolant system leak rate occurred on approximately June 9, 2001.

Although plant operators were successful at identifying an increasing trend in the unidentified primary coolant system leak rate at an early stage (approximately 0.1 gpm), plant personnel did not quantify the contribution to the total leakage from the pressurizer sample line swagelock fitting during the time of plant shutdown. The licensee performed a leakage calculation to evaluate whether leakage from the crack could reasonably account for containment sump in-leakage prior to shutdown. However, due to the uncertainty associated with calculating leakage from the small flaw in the relatively large housing, combined with lack of leak rate data from the pressurizer sample line swagelock fitting, plant staff could not accurately quantify the leak from the CRD-21 housing. The licensee determined, based on visual observation, that the containment sump in-leakage was a combination of the pressurizer swagelock fitting leak and the CRD-21 housing leak.

Although the licensee did not have data on the magnitude of the individual leaks, at the time of plant shutdown, the unidentified primary coolant system leak rate was approximately 0.3 gpm, within the Technical Specification 3.4.13 limit of 1.0 gpm. However, Technical Specification 3.14.3 requires that primary coolant operational leakage shall be limited to "No pressure boundary LEAKAGE" when in Modes 1-4. The associated action requires that the plant be placed in Hot Standby (Mode 3) within 6 hours and in Cold shutdown (Mode 5) within the following 36 hours. Although the time the pressure boundary leakage began could not be precisely determined, it is clear that the leakage existed greater than the 6 hours that was required for the plant to be in Mode 3. Pending further review of the circumstances of this potential violation and the application of the NRC's Enforcement Policy, this will be an Unresolved Item (URI 50-255/01-11-01).

.2 Determination of Root Cause for Control Rod Drive Housing Cracking

a. Inspection Scope

The inspectors reviewed the licensee's preliminary investigation and root cause documents and interviewed members of the root cause investigation team.

b. Findings

Non-destructive examination revealed that an axially-orientated flaw had propagated from the inside surface of the eccentric reducer to pipe weld on the housing of CRD-21. On June 27 and July 1, 2001, during examinations for extent of condition, plant personnel identified indications on CRD-25 and -40. The licensee's root cause investigation team determined that the likely cause of the cracking identified in the housings of CRD-21, -25 and -40 was chloride-induced transgranular stress corrosion cracking. This conclusion was based on several factors including:

- material fabrication history;
- environmental conditions (stagnant environment);
- stress conditions;
- initiation site;
- industry operating experience; and
- previous station operating history.

The licensee staff reported that the root cause was supported by the metallographic examination of the cracks examined from the removed section of the CRD-21 housing at the pipe to eccentric reducer butt-weld (termed "Weld 3"). The metallurgical report was still in draft and not available for review at the conclusion of this inspection.

.3 Safety Significance

f. Inspection Scope

The inspectors reviewed the safety significance associated with the leaking/cracked CRD housings, and the potential for catastrophic rupture. In addition, the inspectors reviewed the safety significance associated with boric acid contact on reactor vessel head components.

g. Findings

b.1 Control Rod Drive Housing Crack and Potential Catastrophic Failure

For this specific event, the reactor was shut down and placed in Cold Shutdown (Mode 5) with low primary coolant system leakage. The axial orientation of the eccentric reducer to pipe weld crack on the housing of CRD-21 resulted in a slowly increasing rate of primary coolant leakage which was monitored by plant operators. The slowly increasing leakage and also provided operators sufficient time to place the reactor in a safe shutdown condition. Because the leak was small and all accident mitigation equipment was available, the shutdown resulted in a conditional core damage probability of 1.3E-06, which was not considered to be a risk significant event.

Regarding the risk evaluation for a postulated catastrophic failure of the CRD housing, assuming the primary coolant leak area was the entire diameter of the CRD assembly housing throat, the catastrophic failure of the CRD housing most closely corresponded to a medium break loss of coolant accident with a conditional core damage probability of 1.8E-02. If the CRD rack remained in the throat of the CRD housing (control rod not ejected), or the break was partial or not catastrophic, the leak could possibly be the size of a small break loss of coolant accident with a conditional core damage probability of 1.18E-02.

Although the size of the leak from the crack in CRD-21 was not outside the bounds of the plant licensing basis, given the amount of cracking that had been discovered to date, it is unclear how the initiating event probability and core damage frequency had been changed. In addition, the impact on risk analysis of several potential engineering concerns documented in later sections of this report has not yet been evaluated.

b.2 Boric Acid Leakage on Reactor Vessel Components

During initial discussions with the licensee the inspectors asked questions regarding the amount and impact of boric acid leakage onto the reactor vessel head components that plant personnel subsequently addressed.

The licensee performed inspections for boric acid during the current forced outage and identified no evidence of primary coolant system leakage from any source on or near the reactor vessel head except for the leakage from the housing of CRD-21. On June 24, 2001, the licensee removed the stainless steel insulation cover panels in the area of the CRD-21 housing leak. The exposed area (CRD mechanism nozzles down to the insulation blankets and upper surfaces of exposed blankets) was visually examined and digital photographs were taken. The licensee found no boric acid liquid and the licensee reported the insulation blankets to be dry. The only leakage was noted to be above the stainless steel insulation cover panels due to spray from CRD-21 which was subsequently removed.

On June 26, 2001, the licensee visually re-examined the exposed insulation blankets, and inserted a lighted fiberscope at various points immediately adjacent to the CRD-13 and CRD-21 nozzles where the nozzles penetrate the insulation. The licensee examined CRD-13 because it was adjacent to and received the spray from the CRD-21 housing leak. The licensee found no evidence of wetness or boric acid adhesion to metal surfaces below the insulation at CRD-13 and CRD-21. Boric acid residue that sprayed from the CRD-21 housing leak remained on the insulation blankets and had dried out and was removed by vacuuming.

The licensee performed a bare metal inspection of the reactor vessel head in 1995 and installed new insulation blankets. During the bare metal inspection the licensee identified no evidence of boric acid leakage at penetrations into the reactor vessel head.

.4 Extent of Condition Evaluation

a. Inspection Scope

The inspectors reviewed the licensee's activities initiated to determine the extent of the CRD housing cracking.

b. Findings

Over the service life of the plant the licensee had performed only external surface examinations of a sample (10 percent) of the peripheral CRD mechanism housing welds as allowed by the American Society of Mechanical Engineers (ASME) Code for Code Category B-O "Pressure Retaining Welds in Control Rod Housings." Following the discovery of the crack in CRD-21, the licensee initially performed dye penetrant examinations of the external surface of welds on 5 CRD housings located on the periphery of the vessel.

To assess the extent of condition from inside diameter cracking, the licensee performed the following types of non-destructive examinations:

- Housing Weld 1 (first weld above the vessel head, nozzle to flange weld) - Ultrasonic examinations conducted from the pipe side of this pipe to flange weld for 26 CRD housing locations accessible from the periphery of the vessel head.

- Housing Weld 2 (second weld above the vessel head, flange to eccentric reducer weld) - Ultrasonic and radiographic examinations of 3 CRD housings and one inside diameter dye penetrant examination on removed CRD-21 housing.

- Housing Weld 3 (third weld above the vessel head, eccentric reducer to pipe weld) - Ultrasonic and radiographic examinations of 26 CRD housing locations accessible from the periphery of the vessel head. Visual examinations of the inside surface were initiated using a remote camera system on the inside of the CRD housing.

- Housing Weld 4 (fourth weld above the vessel head, pipe to flange weld) - Dye penetrant examinations conducted from the inside diameter at 3 CRD housing locations.

- Housing internal weld buildup location - Ultrasonic examinations of 26 housings locations accessible from the periphery of the vessel head.

At the conclusion of this inspection the licensee had identified axial and circumferential crack indications in multiple housings associated with Weld 3 and one indication in the counterbore near Weld 3 of the removed CRD-21 housing. Enclosure 1 presents the results of the examination of the section removed from the housing of CRD-21, which was removed to confirm the root cause of the cracking. Further planned non-destructive examination included visual examination of the inside diameter surface of Weld 3 for all CRD housings using a remote controlled camera system.

The licensee formed a root cause investigation team, which was tasked with evaluating the extent of condition associated with the CRD housing cracks and identifying other plant components susceptible to cracking. The root cause team developed a table of plant components and locations which were potentially susceptible to cracking. This table and evaluation of the plant components susceptible to cracking was not complete at the conclusion of this inspection. The licensee staff indicated that no other Code Category weldments (other than the CRD housings) would require additional non-destructive examinations prior to plant restart.

The previous station operating history included cracking in weld heat affected zones of CRD mechanism seal housings fabricated from type 347 and 304 austenitic stainless steel. The discovery of stress corrosion cracking raises a question regarding the adequacy of actions to evaluate the extent of condition and corrective actions from prior cracking in the control rod drive seal housings discovered during the 2001 refueling outage. The evaluation to be performed by the licensee's root cause team included reviewing corrective actions taken for CRD housing cracks discovered in prior Palisades outages. Pending NRC review of the prior corrective actions for CRD housing cracks, this issue is considered an Unresolved Item (URI 50-255/01-11-02).

.5 Adequacy of Repair Methodology

a. Inspection Scope

The inspectors reviewed the licensee's original proposed design change documented in EAR-2001-0373-01, "Justify Use of Weld Overlay per Code Case N-504-1 for Repair of Control Rod Drive Mechanisms 25 and 40. Justification Will Be Available for Use of the Overlay Technique for Repair of Leaks of Other Control Rod Drive Mechanism Housings."

b. Findings

In EAR-2001-0373-01, the licensee proposed an external weld overlay type repair in accordance with ASME Code Case N-504-1, "Alternate Rules For Repair of Class 1, 2, and 3 Austenitic Stainless Steel Piping, Section XI, Division 1." This overlay design was comprised of at least two weld passes with a minimum weld buildup of 0.125 inches.

The adequacy of the original overlay design was based on the measured crack length from ultrasonic examinations without consideration for uncertainty on the length sizing of the cracks. Further, the ultrasonic examination procedure used (LMT-PDI-UT-2) documented a limitation, in that the procedure was not qualified for length sizing of axial flaws. Radiographic examination of housing cracks on CRD-21 revealed that crack lengths recorded based on ultrasonic examinations were undersized by up to1.0 inch. The undersizing of crack length was attributed to the inability of the ultrasonic examinations to adequately probe into or through the housing weld material.

The adequacy of the original overlay design was based on a crack growth rate of 4.5×10^{-6} inches per hour as documented in EAR-2001-0373-01, which was derived mainly from data associated with intergranular stress corrosion cracking. A technical paper "The Sixth International Symposium on Environmental Degradation of Materials in

Nuclear Power Systems - Water Reactors," presented data from the cracked CRD mechanism housings at the Fort Calhoun station which had a CRD housing design and operating history similar to Palisades. Based on the data from this paper, the inspectors calculated a crack growth rate for transgranular stress corrosion cracking in Fort Calhoun CRD housings, that was approximately three times greater than the crack growth rate proposed by the licensee for the Palisades CRD housing cracks. Pending the NRC's further evaluation of the licensee's basis for the assumed crack growth rate in the initial overlay design, this issue is considered an Unresolved Item (URI-50-255/01-11-03).

.6 Adequacy of Overall Corrective Actions

a. Inspection Scope

The inspectors reviewed the licensee's initial corrective actions and interviewed members of the licensee's root cause team to assess the adequacy of the corrective actions.

b. Findings

b.1 Initial Corrective Actions

The licensee had formed a root cause team to investigate the cause of the CRD cracking and had removed the CRD-21 housing to perform non-destructive and destructive examinations to characterize the cracking and confirm the root cause. The root cause team investigation into the cause of the cracking was continuing at the conclusion of this inspection. The preliminary root cause for the cracking is discussed in Section 40A3.2.

To evaluate the extent of condition for the CRD housing cracks, the licensee performed ultrasonic, radiographic, dye penetrant and visual examinations at housing weld locations identified in Section 40A3.4. Other non-destructive examination techniques, such as eddy current, were under consideration. Additionally, the scope of housing welds subject to other types of non-destructive inspection techniques was also under review and subject to change.

The licensee's short and long term corrective actions were not fully developed at the conclusion of this inspection. The repair options for CRD housings under consideration at the conclusion of this inspection included a full structural overlay of cracked housing welds or replacement of housings. Additionally, the licensee was developing an analytical approach which would allow returning cracked housings to service for crack sizes which could be demonstrated to not challenge the structural integrity of the housing.

b.2 Corrective Action Issues

b.2.1 Unanalyzed Missile Shield Modification

A potential consequence of the CRD housing cracks was the increased probability of a design basis rod ejection accident. During their efforts to evaluate the risk significance, the licensee identified that the support configuration of the missile shield over the reactor vessel, which mitigates the design basis rod ejection accident, was different than that shown in design documents. The licensee initiated C-PAL-01-02248 and performed an operability evaluation since no calculations could be found to justify the as-built configuration. The operability evaluation concluded that the missile shield would perform its design function in the as-built configuration for both a rod ejection accident and a seismic event.

The NRC inspectors reviewed the above operability evaluation and determined that a critical component of the as-built configuration had not be considered. Based on the conservative methodology in the operability evaluation, the inspectors concluded that the capacity of the laterally-loaded 36-inch I-beam in the missile shield support structure would be greatly exceeded during a design basis seismic event. This would allow the missile shield to fall onto the reactor vessel head, damaging the CRDs and preventing the reactor from shutting down. Following discussions with the inspectors, the licensee initiated C-PAL-01-02647 to document the inadequacies in their initial operability evaluation and declared the missile shield inoperable pending further evaluation. The significance of this issue can not be determined until the licensee completes their further evaluation, so this is considered an Unresolved Item (URI 50-255/01-11-04).

b.2.2 Design Basis Loading for CRD Housings

The design basis loading for the CRD housings did not appear well understood by the licensee. The loads on the housings are crucial in determining the critical crack size for the CRD housings and in evaluating the adequacy of the modification for the proposed replacement housing on CRD-21.

According to the licensee, the design basis calculation for the CRD housing is a Combustion Engineering Report No. TR-ESE-437, “Palisades CRDM Dynamic Analysis Report,” July 6, 1981. However, because of the lack of details in the report, the licensee chose to use the information from an initial analysis of the CRD housing done in 1967. Although the initial analysis appeared bounding with respect to the applied bending moments, it did not include an 18,000 pound axial force on the CRD housing indicated in the 1981 design basis analysis. As a result, the applied loads appear to be non-conservative in the critical crack size calculation, EA-EAR-2001-0373-04, Attachment 1, “Evaluation of Leakage from Circumferential and Axial Through-Wall Cracks in Lower CRDM Housing.”

The above axial load was also not considered in modification EAR-2001-0382, “CRD Upper Housing Replacement.” The modification package stated that the seismic restraint collar, originally provided by a tapered 1/4-inch weld build-up pad on the outside diameter, only facilitated the installation of the seismic restraint. The modification package further stated that replacing the collar design with two rings, attached to the housing using 1/8-inch skip welds, had no impact on the structural acceptability of the housing. Because it appears that the seismic restraint collar must resist a substantial axial load, this statement may not be true and the design may be inadequate.

In addition to the above concerns, FSAR Section 3.2.3, Design Limits for Control Rods states that, for pipe rupture accident loads, the CRD housing is designed to support and maintain the position of the control rod. The effect of the pipe rupture loads on the CRD housings was reviewed by the NRC in “Safety Evaluation on Asymmetric LOCA [Loss of Coolant Accident] Loads,” dated October 27, 1989. This review documented an applied bending moment to the CRD nozzle as 142,000 inch-pounds, which is substantially higher than the 75,000 inch-pounds documented in the previous 1967 analysis. However, the need to apply asymmetric LOCA loads to the CRD housing may not be required due to the application of leak-before-break methodology, but the licensee could not provide documentation to support this aspect. Pending verification of the design basis load requirements by the licensee, this will be considered an Unresolved Item (URI 50-255/01-11-05).

b.2.3 <u>Leak Rate Flow Effect on Rod Function</u>

The licensee did not consider the effect that leakage flow would have on the function of the control rod. In determining the critical crack size for the CRD housing, the evaluation only considered the structural stability from a fracture mechanics perspective and did not consider the potential consequence of leakage flow up through the CRD nozzle on the function of the control rod.

Calculation EA-EAR-2001-0373-01, Attachment 4, “Safety Assessment Report for the Palisades Nuclear Plant Control Rod Drive Mechanism Weld Overlay,” Section 3.0 discusses the CRD functions. It states that the CRD housing is a passive component whose sole purpose is to retain structural integrity and thus not interfere with movement of the operating mechanisms contained within the boundary. It goes on to state that if the flaw is below the critical flaw size, then failure of this location will not occur and distortion will be minimal, allowing the component to maintain its intended function. The above discussion fails to recognize that, while the housing may be structurally adequate, the CRD may not maintain its intended function if the leak rate through a crack is excessive. An excessive leak rate could prevent the control rod from scramming as a result of the high differential pressure between the reactor and the housing due to the flow restrictions through the CRD nozzle. The flow rate calculation for this purpose would require assumptions to maximize instead of minimizing the leakage flow rates for a given crack size. The licensee’s subsequent calculation to address this issue did not clearly demonstrate that the critical crack size would not be affected by this issue. Pending additional evaluation of the leak rate from the critical crack size on control rod function, this will be considered an Unresolved Item (URI 50-255/01-11-06).

b.2.4 <u>Rod Ejection Effect on Adjacent Rods Due to Seismic Restraint</u>

The original design basis of the CRD housing’s seismic restraint did not include the effects of a rod ejection accident on the function of the adjacent control rods. The issue is whether the CRDs adjacent to an ejected CRD housing would be able to perform their function due to the significant forces and moments transferred through the CRD seismic restraints.

In determining the potential significance of the CRD housing cracks, the licensee considered the potential beneficial effects of the CRD seismic restraints for a design basis rod ejection accident. Based on a conservative evaluation, the licensee concluded that the bolts in the seismic restraints would be over stressed for the peripheral CRD housings. Reviews of the CRD seismic restraint design basis concluded that rod ejection loads were not considered and the bolts being over stressed appeared to be a beyond-design-basis situation.

However, while the seismic restraints are not explicitly designed to restrain a housing during a rod ejection accident, they will restrain an ejected CRD housing to some extent. In doing so, the adjacent CRD housings will have significant forces and moments imposed upon them, for which they were not designed, but potentially should have been. These loads may prevent the adjacent CRDs from being able to scram, which did not appear to have been considered for this design basis accident. This appears to be a potential initial design inadequacy. Pending a review of the licensee's evaluation of a rod ejection's effect on the adjacent control rods, this will be considered an Unresolved Item (URI 50-255/01-11-07).

.7 <u>Similarity with Other Leakage Issues</u>

a. <u>Inspection Scope</u>

The inspectors reviewed the scope of the licensee's root cause team investigation to assess the use of industry leakage events.

b. <u>Findings</u>

The scope of the extent of condition evaluation to be performed by the root cause team included reviewing actions taken for prior instances of CRD cracks throughout the industry. Examples included the reactor vessel head CRD stub tube cracking at Oconee and the cracking in the spare control rod drive mechanism housing in the weld buildup area experienced at Fort Calhoun. This evaluation was not complete at the conclusion of this inspection.

.8 <u>Quality of Non-destructive Examination</u>

a. <u>Inspection Scope</u>

The inspectors reviewed radiographic records, ultrasonic and dye penetrant examination records, and interviewed non-destructive examination personnel to assess the quality of the non-destructive examinations of CRD housing welds.

b. Findings

The ultrasonic examinations of CRD housing welds were conducted with contract non-destructive examination personnel trained and qualified to Appendix VII of Section XI of the ASME Code. The ultrasonic examination procedure utilized was demonstrated by performance testing to be effective at detecting cracking in austenitic pipe welds in accordance with Appendix VIII of Section XI of the ASME Code. These examinations successfully detected axial cracking in the CRD mechanism housings at Weld 3 (eccentric reducer to pipe weld). However, the ultrasonic examinations failed to detect a circumferential crack over 90 percent through wall which initiated at the counterbore below Weld 3 for CRD-21. Further, ultrasonic examinations failed to adequately detect the length of deep axial cracks within the weld material and failed to detect shallower circumferential and axial cracks in the weld and base material on the removed section of CRD-21 housing.

Ultrasonic examinations were conducted on Weld 1 (flange to pipe weld) on the 26 CRD housings accessible from the periphery of the reactor vessel head. The axial scans for these ultrasonic examinations were limited by surface geometry such that scans were conducted from the pipe side of the weld. This examination would not likely detect potential circumferential cracks in the weld metal or flange side base metal on Weld 1.

The licensee used double wall radiography to supplement ultrasonic examinations conducted at the Weld 3 location for 26 housing locations and at the Weld 2 location for three housing locations on the periphery of the vessel head. The radiographic examination records were annotated as “information only.” The inspectors noted that the 2T hole was visible on the penetrameter in the radiographs of housing welds indicating that image quality was sufficient to meet Code requirements. The radiographic examinations detected more indications in the weldments than were detected by ultrasonic examinations. Further, based on destructive examinations, the radiographic examinations provided potentially more accurate information for determining the lengths of deeper cracks. However, radiographic examinations failed to detect the shallower axial and circumferential cracks identified in the removed section of CRD-21 housing.

Based on the data in Enclosure 1 taken from CRD-21, -25 and -40, ultrasonic inspections were successful at detecting axial cracking with substantive through wall extent in the base metal of the housing welds if no geometric challenges exist. However, failure to detect the cracking which progressed into weld material indicated that the ultrasonic examinations could not reliably detect cracking contained entirely in the weld material. Further, neither radiographic nor ultrasonic examinations could detect shallow cracking on the inner diameter of the CRD mechanism housing. For these reasons, and to examine housings inaccessible for ultrasonic/radiographic examination, the licensee initiated inside diameter visual examinations of Weld 3 utilizing a remote camera system.

.9 Adequacy of Radiological Controls

a. Inspection Scope

The inspectors performed on-site inspections of the licensee's As-Low-As-Reasonbly-Achievable (ALARA) practices, including administrative, operational, and engineering controls. The inspectors performed the inspections in early June 2001 and again in July 2001 as work activities on the CRD housing project evolved.

b. Findings

The inspectors compared the ALARA work plans with the results achieved and determined that the results achieved were consistent with the plans. The inspectors reviewed radiation work permits for various work activities and noted good integration of ALARA requirements into the radiation work permits. The inspectors evaluated the interfaces between operations, radiation protection, maintenance, maintenance planning, scheduling and engineering groups and noted good communications between the groups. The inspectors also reviewed the ALARA group's temporary shielding packages and noted significant reductions in dose rates around the head and the targeted CRD mechanism housings as well as timely and well coordinated engineering support for the ALARA group's temporary shielding requests. The inspectors noted that ALARA in-progress reviews had been conducted in response to changes in the scope of work activities and ALARA related problems identified during the work activities had been entered into the licensee's corrective action program.

No findings of significance were identified.

4OA6 Meeting(s)

Exit Meeting

The inspectors presented the inspection results to Mr. Cooper and other members of licensee management at the conclusion of the inspection on August 9, 2001. The licensee acknowledged the findings presented. Proprietary information was discussed with the licensee and appropriately handled.

Enclosure: As Stated

Palisades CRD 21 Housing Indications					
Indication/Location *	Information	Exterior UT	Interior PT	Single Wall Radiography	Double Wall Radiography
1 11:30 - 1:00	Original through wall Indication which extended through weld and base metal on both sides of Weld 3.	Identified	Identified (2.5 inches long)	Identified (2.25 inches long)	Identified (2.25 inches long)
2 11:00 - 11:30	Axial with circumferentially oriented branch indication in Weld 3 (0.1875 inches deep)	Not Identified	Identified (1.25 inches long)	Not Identified	Not Identified
3 11:00	"Z" shaped Indication in Weld 3.	Not Identified	Identified (1.25 inches long)	Not Identified	Not Identified
4 5:00 - 7:00	Circumferential Indication on the counterbore radius of the eccentric reducer which was 75 percent through wall.	Not Identified	Identified (2.5 inches long)	Identified (2.125 inches long)	Identified (2.0 inches long)
5 4:00	Circumferential Indication in Weld 3 not connected to inside or outside surface.	Not Identified	Not Identified	Identified (0.375 inch long)	Not Identified

* Location relative to 12:00 being center of "flat" section of eccentric reducer

Palisades CRD 25 housing indication					
Indication	Information	Exterior UT	Interior PT	Single Wall Radiography	Double Wall Radiography
1	Axial Crack in Weld 3	Identified (0.5 inches long)		NA	Identified 1.5 inches long

Palisades CRD 40 housing Indication					
Indication	Information	Exterior UT	Interior PT	Single Wall Radiography	Double Wall Radiography
1	Axial Crack in Weld 3	Identified (0.7 inches long)		NA	Identified on second set of radiographs (0.6 inch long) after first set failed to identify this indication.

KEY POINTS OF CONTACT

<u>Licensee</u>

D. Cooper, Site Vice President
B. Dotson, Licensing Analyst
B. Gerling, Licensing Support Supervisor
P. Harden, Director, Engineering
D. Malone, Acting Director, Licensing and Performance Assessment
T. Kirwin, Plant General Manager (Acting)
M. Carlson, Engineering Programs Manager
G. Goralski, Design Engineering Manager
L. Ross, Planning and Scheduling Manager
J. Hager, Engineering Programs
S. Wawro, Consumers Energy/Asset Manager
T. Fouty, Engineering Programs ISI
B. VanWagner, Design Engineering

<u>NRC</u>

J. Grobe, Director, DRS
A. Vegel, Chief, Branch 6, DRP
D. Passehl, Project Engineer, DRP
M. Holmberg, Mechanical Engineering Branch, DRS
J. Gavula, Mechanical Engineering Branch, DRS
J. Lennartz, SRI, Palisades

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

50-255/01-11-01	URI	NRC Review of Technical Specification Pressure Boundary Leakage Relative to Enforcement Policy
50-255/01-11-02	URI	NRC Review of the Prior Corrective Actions for Control Rod Drive Housing Cracks
50-255/01-11-03	URI	NRC Review of Licensee's Basis for Use of Crack Growth Rate in the Initial Weld Overlay Design
50-255/01-11-04	URI	NRC Review of the Operability Evaluation for the Unanalyzed Missile Shield Modification
50-255/01-11-05	URI	NRC Review of the Design Basis Loading for CRD Housings for Critical Crack Size and Replacement Housing Modification
50-255/01-11-06	URI	NRC Review of the Flow Effect from the Critical Crack Leak Rate on Control Rod Function
50-255/01-11-07	URI	NRC Review of Rod Ejection Effect on Adjacent Rods Due to Seismic Restraint

LIST OF ACRONYMS USED

ALARA	As-Low-As-Reasonably-Achievable
ASME	American Society of Mechanical Engineers
CRD	Control Rod Drive
FSAR	Final Safety Analysis Report
LOCA	Loss of Coolant Accident

LIST OF DOCUMENTS REVIEWED

Procedures

GOP 13	Primary Coolant System Leak Rate Calculation	Revision 13
LMT-PDI-UT-2	Ultrasonic Examination of Austenitic Piping Welds	Revision 0

Work Requests

WO24111543	Replace Pressurizer Vent Line and Fittings	

Radiation Work Permits and Associated ALARA Job Reviews

PO11008	Electrical & Mechanical Maintenance to Remove various CRDM drive packages and seal housings.	Revision 0
PO11011	Control Rod Drives #'s 21, 25 and 40 Pressure Boundary Support Tube Project which includes:	Revision 4
PO11012	Installation and removal of various lead shielding packages around the Reactor Head and Cavity which includes post shielding surveys.	Revision 3
PO11014	Non-Destructive Testing for Control Rod Drive Pressure Boundary Support Tube.	Revision 2
PO11015	Decon of the area of leakage near support tube number 21 and under the insulation if needed.	Revision 1
PO11016	Support activities for Control Rod Drive repairs which include:	Revision 0
PO11019	Removal of CRD-21 support housing and install new upper support housing.	Revision 0

Temporary Shielding Requests

2001-60	649 foot CTMT and Rx. Cavity Floor	June 21, 2001
2001-61	CRDM Stalk Shielding	June 21, 2001
2001-62	ICI Flange	June 21, 2001
2001-63	Install Additional Shielding Floor of Reactor Cavity	July 6, 2001
2001-64	Add Additional Shielding on Scaffolding	July 3, 2001

Condition Reports (CPAL)

CPAL0101851	Swagelock Leak Downstream of MV-PC1045C PCS PZR Vapor Sample Line
CPAL 0102186	Primary Coolant System Pressure Boundary Leakage CRD-21 Support Tube

CPAL0102244	TSR 2001-64 for CRD-21 Support Tube Housing Grinding Work Did Not Accurately Reflect as Built Scaffolding Dimensions Due to Communication Errors
CPAL0102247	Carbon Steel Fasteners Exposed to Borated Water on CRD-13
CPAL0102248	Missile Shield over Reactor Vessel Is Support Different than Shown in Design Documents
CPAL0102253	Control Rod Drive Support Tube Project RWP Not Revised to Support Scheduled Work
CPAL0102257	Axial Indication Identified on CRD Support Tube for CRD-40
CPAL0102258	Unable to Locate Original Fabrication Records for CRDM'S
CPAL0102261	Capscrews on Seismic Restraint Won't Come Loose
CPAL0102262	Bolt on the Seismic Bar and Cylinder Will Not Come Loose
CPAL0102272	FSAR Terminology for CRDM Upper Housing Assembly Does Not Match Drawings
CPAL0102274	Scaffold Height Inadequate for CRD-21 Bevel Gear Housing Grinding Evolution
CPAL0102281	CRD-21 Support Tube Internals Visual Inspection Results
CPAL0102282	Exact Location of Exit Point of Through Wall Flaw in CRD-21
CPAL0102283	Flapper Wheeling on CRD-21 Support Tube Appeared More Difficult in the Field Then on the Mock-up
CPAL0102284	11/16" Crack Protruding into Weld Face of CRD-21 Discovered
CPAL0102291	Option Provided to Field for Repair of CRD Should Not Have Been Provided
CPAL0102356	PCS DBD Statement of the CRD Seismic Restraints Serve to Prevent Rod Ejection Has No Basis
CPAL0102358	Inadequate Posting Following Cut up of CRD-21 Housing

CPAL0102372	Additional Dose Received During Installation of "Top Hat" on CRD-21 Nozzle	
CPAL0102647	Inadequacies Identified in Operability Evaluation for CPAL0102248	
CPAL0102649	Corrective Action Not Identified for Apparent Cause	

<u>Radiographic Records</u>

Weld P-224 (essentric reducer to pipe weld #3)	Package #19 (CRDM 40)	dated 8/07/1968
Weld P-224 (essentric reducer to pipe weld #3)	Package #26 (CRDM 21)	dated 8/16/1968
Weld P-224 (essentric reducer to pipe weld #3)	Package #44 (CRDM 25)	dated 11/21/1968
Weld P-224 (essentric reducer to pipe weld #3)	Information Radiographs on Removed Section of CRD-21 Housing	not dated
Weld P-224 (essentric reducer to pipe weld #3)	Project 0100409 (CRDM-25)	dated 7/14/2001
Weld P-224 (essentric reducer to pipe weld #3)	Project 0100409 (CRDM-40) (2 packages)	dated 7/15/2001 and 7/16/2001

<u>Ultrasonic Examination Records</u>

01-001	Weld 119-33D	dated 6/24/2001
01-002	Weld buildup CRD 33, 38, 40, 41, 44,	dated 6/28/2001
01-003	Welds 119-33B, 41B, 40B, 38B	dated 6/28/2001
01-004	Weld 119-44B	dated 6/28/2001
01-005	Welds 119-33E, 41E, 40E, 39E, 44E, 44D	dated 6/28/2001
01-007	Weld 119-31D	dated 6/28/2001
01-008	Welds 119-43B&C, 35B&C, 29B&C, 45B&C, 37C, 26C, 30B&C, 39B&C, 31B&C, 27B&C, 32B&C, 28E,34E, 42E, 43E, 35E, 29E, 36E, 45E, 37E, 26E, 30E, 39E, 31E, 27E, 32E	dated 6/30/2001
01-009	Welds 119-22B, 22C, 22D,23B, 23C, 23D	dated 7/01/2001

Drawings

CND-SD-2001	Bottom Flange	Revision D
CND-SD-2428	Upper Flange	Revision 1
CND-SD-2427	Pipe Weld Buildup and Machining	Revision 3
CND-SD-1794	Modified Eccentric Reducer	Revision B
CND-SE-2689	Upper Housing Weldment	Revision A
CND-SE-2175	Upper Housing Assembly	Revision Y
232-119-11	Closure Head Nozzle Details	Revision 11

Purchase Orders

980046	Upper Housing Assembly Bottom Flange	1/26/1968
9701365	8X5 Schedule 160 Eccentric Reducers Type 347 Stainless Steel	11/21/1967
9701365	8X5 Schedule 160 Eccentric Reducers Type 304 Stainless Steel	11/3/1967
9701329	8 inch Schedule 120 12 foot 4 inch Long Pipe Type 304 Stainless Steel	11/3/1967
9701329	8 inch Schedule 120 12 foot 4 inch Long Pipe Type 347 Stainless Steel	11/21/1967

Material Specifications

The Timken Roller Bearing Company 9701329 Heat 22574	Electric Furnace TP-347 Cold Finished-Solution Treated-Pickled Spec ASTM-A-312 Grade TP-347	March 4, 1968
The Timken Roller Bearing Company 9701329 Heat 22577	Electric Furnace TP-347 Cold Finished-Solution Treated-Pickled Spec ASTM-A-312 Grade TP-347	March 4, 1968
Pennsylvania Forge, A15373, Heat 68192	Essentric Reducer, ASTM A182 Gr F347	February 16, 1968

Modifications/Calculations

EAR-2001-0373	Evaluate Weld Overlay as a Permanent Repair Option for CRDM-21	Revision 0
EAR-2001-0373-01	Justify Use of Weld overlay per Code Case N-504-1 for Repair of CRD-25 and CRD-40. Justification will be Available for Use of the Overlay Technique for Repair of Leaks of Other CRD Housings.	Revision 2
EAR-2001-0373-01,	Attachment 4, Safety Assessment Report for the Palisades Nuclear Plant Control Rod Drive Mechanism Weld Overlay	Revision 2
EAR-2001-0373-04,	Attachment 1, Evaluation of Leakage from Circumferential and Axial Through-Wall Cracks in Lower CRDM Housing.	Revision 1
EAR-2001-0382,	CRD Upper Housing Replacement	Revision 0
TR-ESE-437	Combustion Engineering Report, Palisades CRDM Dynamic Analysis Report	July 6, 1981

Other Documents

Primary Coolant System Trend Charts		May 4, 2001, to July 3, 2001
Shift Supervisor Logs	Various	
EA-PSA-2001-025	Risk Significance of Shutdown Due to CRD 21 PCS Leak	Revision 0
Technical Paper	Sixth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, San Diego CA	August 1-5, 1993