March 4, 2002

EA-02-011

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 50-255/01-15

Dear Mr. Cooper:

On January 29, 2002, the NRC completed 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 January 29, 2002, with you and other members of your staff.

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, a member of your staff and a resident inspector identified an active steam/primary coolant system leak from Control Rod Drive Mechanism (CRDM) 21. Your staff subsequently placed the plant in cold shutdown and assembled a team to evaluate the occurrence, assess extent of condition, determine root cause, and develop strategies for restoration. Non-destructive examination revealed an axially-orientated flaw at the eccentric reducer to pipe weld on the housing of CRDM 21. During subsequent examinations, indications of cracks were identified on multiple additional control rod drive mechanism housings caused by transgranular stress corrosion cracking. To correct this condition, your staff replaced all 45 CRDM housings.

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. Seven violations of regulatory requirements were identified during this inspection. The root cause of these violations appears to include contributions from human performance errors.

Deficiencies in human performance can impact multiple reactor safety cornerstones and cause an adverse impact on safe plant operation. As such, human performance represents a fundamental underpinning of the reactor safety cornerstones. A significant cross-cutting human performance finding was identified associated with these violations in that multiple examples of inadequate engineering products were identified which provided a technical bases for modifications, operability evaluations and corrective actions. This performance deficiency reflects a lack of rigor applied to performing and verifying mechanical, structural, and metallurgical engineering work that affected the reactor safety cornerstones for initiating event frequency, barrier integrity, and mitigating systems. We are concerned that without NRC intervention, inadequate modifications may have been installed and/or that degraded equipment would have been returned to service without an adequate basis to confirm operability. Your on-going effort to perform a comprehensive review to identify and correct the causes of this adverse trend in human performance is important to ensure the integrity of the plant design basis.

The seven violations are categorized as “Green” findings of very low safety significance. Because of their very low safety significance and because these issues were entered into your corrective action program, the NRC is treating these issues as Non-Cited Violations (NCVs) in accordance with Section VI.A.1 of the NRC’s Enforcement Policy. If you deny these Non-Cited Violations, you should provide a response with a basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Palisades Nuclear Generating Plant.

An additional violation of Technical Specification 3.4.13 was identified for plant operation greater than 6 hours with leakage from the cracked CRDM 21 housing. Although this issue constitutes a violation of NRC requirements, it did not have actual safety consequences (as defined in Section IV.A.5.c of the Enforcement Policy), impede the regulatory process, or result from willful acts. Additionally, your staff’s actions did not contribute to the degraded condition and the leakage resulted from material failure not avoidable through reasonable quality assurance measures or management controls. Consequently, no performance deficiency was identified. Based on these facts, the NRC has decided to exercise enforcement discretion in accordance with Section VII.B.6 of the Enforcement Policy and refrain from issuing enforcement action for the violation. A deterministic risk assessment considering fracture mechanics structural margins determined that this was an issue of very low safety significance.

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,

/RA/

John A. Grobe, Director
Division of Reactor Safety

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

Enclosure: Inspection Report 50-255/01-15(DRS)

cc w/encl: R. Fenech, Senior Vice President, Nuclear
Fossil and Hydro Operations
L. Lahti, Manager, Licensing
R. Anderson, Chief Nuclear Officer, NMC
A. Udrys, Esquire, Consumers Energy Company
S. Wawro, Nuclear Asset Director, Consumers Energy Company
W. Rendell, Supervisor, Covert Township
Office of the Governor
Michigan Department of Environmental Quality
Department of Attorney General (MI)

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Michigan Department of Environmental Quality
Department of Attorney General (MI)

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-255
License No:	DPR-20
Report No:	50-255/01-15(DRS)
Licensee:	Nuclear Management Company, LLC (NMC)
Facility:	Palisades Nuclear Generating Plant
Location:	27780 Blue Star Memorial Highway Covert, MI 49043-9530
Dates:	November 19, 2001 through January 29, 2002
Inspectors:	Melvin Holmberg, Inspector James Gavula, Inspector
Approved by:	John Jacobson, Chief Mechanical Engineering Branch Division of Reactor Safety

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SUMMARY OF FINDINGS

IR 05000255-01-15(DRS), on 11/19/2001-01/29/2002, Nuclear Management Company, LLC, Palisades Nuclear Plant. Special Inspection.

This Special Inspection was initiated to evaluate the facts, circumstances and corrective actions surrounding discovery of a through-wall crack in the control rod drive mechanism 21 housing. This phase of the Special Inspection was conducted by two Region based inspectors and eight findings were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process" (SDP). 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. Findings for which the SDP does not apply are indicated by "No Color" or by the severity level of the applicable violations.

Cornerstones: Initiating Events, Barrier Integrity, and Mitigating Systems

- No Color. A significant cross-cutting human performance finding was identified associated with multiple examples of inadequate engineering products that provided a technical bases for modifications, operability evaluations and corrective actions. This performance deficiency reflected a lack of rigor applied to performing and verifying mechanical, structural, and metallurgical engineering products that affected the reactor safety cornerstones for initiating event frequency, barrier integrity, and mitigating systems.

  The NRC is concerned that without inspector intervention, inadequate modifications would have been installed and that degraded equipment would have been returned to service without an adequate basis to confirm operability. At the conclusion of this inspection, the licensee was in the process of performing a comprehensive review to identify and correct the causes of this adverse trend in human performance (Section 4OA4).

Cornerstones: Barrier Integrity and Initiating Events

- Green. A Non-Cited Violation of Technical Specification 3.4.13 was identified for operation of the plant with pressure boundary leakage from a through-wall crack in the control rod drive 22 seal housing. Although the time that the pressure boundary leakage from control rod drive 22 housing began could not be precisely determined, it is clear that leakage existed for greater than the 6 hour time limit to place the plant in Mode 3.

  This self-revealing finding affected the barrier integrity and initiating events cornerstones and was greater than minor because, if left uncorrected, it could have resulted in further degradation of the reactor coolant pressure boundary. Based on insights from fracture mechanics and leak-before-break perspectives, operation with this degraded CRDM seal housing would not substantially increase initiating event frequency. Therefore the risk significance was very low (Section 4OA3.4).

- Green. A Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” was identified for failure to implement corrective actions to prevent recurrence of cracking identified in the type 347 stainless steel control rod drive seal housings. Following identification of a through-wall crack, the service life of the housings was in question and no service life demonstration was completed before restart of the plant.

  This finding had the potential to affect the barrier integrity and initiating events cornerstones and was greater than minor because, if left uncorrected, it could have resulted in breach of the reactor coolant pressure boundary. Fortuitously, the licensee operated for only 1 month after placing these housings back in service. Therefore, no actual degradation of the primary pressure boundary occurred and the risk significance was very low (Green) as determined by the Reactor Safety Significance Determination Process (Section 4OA3.4).

  A violation of Technical Specification 3.4.13 was identified for plant operation greater than 6 hours with leakage from the cracked CRDM 21 housing. Although this issue constituted a violation of NRC requirements, the entry conditions for evaluation under the traditional enforcement program were not satisfied, in that, this issue did not have actual consequences (as defined in Section IV.A.5.c of the Enforcement Policy), impede the regulatory process, or result from willful acts. Additionally, this issue was evaluated under the Reactor Oversight Process. The NRC concluded that the licensee’s actions did not contribute to the degraded condition and, thus, no performance deficiency was identified. Based on these facts, the NRC has decided to exercise enforcement discretion in accordance with Section VII.B.6 of the Enforcement Policy and refrain from issuing enforcement action for the violation. Because there were substantial structural margins when comparing the critical flaw sizes to the existing flaw sizes, an increase in initiating event frequency for loss of coolant accidents was judged to be unwarranted. Further, the calculated leakage at one-half the critical flaw size would require a plant shutdown, well before reaching a flaw size that would result in a catastrophic housing failure. Therefore, this finding was determined to be of very low risk significance (Section 4OA3.4).

- Green. A Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” was identified for use of a nonconservative crack growth rate in a calculation supporting a weld overlay repair design change for the control rod drive mechanism housings.

  This finding had the potential to affect the barrier integrity and initiating events cornerstones and was greater than minor because, if left uncorrected, it could have resulted in installation of an inadequate overlay repair for a degraded the primary coolant pressure boundary. Subsequently, the control rod drive housings were replaced and the weld overlay design change was not implemented. Therefore, the integrity of the primary coolant system boundary was not affected and this finding was determined to be of very low risk significance (Green) by the Reactor Safety Significance Determination Process (Section 4OA3.5).

- Green. A Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” was identified for failure to consider bending loads at the I-beam’s web in the initial operability evaluation of the discrepant missile shield support structure.

This finding had the potential to affect the barrier integrity and initiating events cornerstones and was greater than minor because it had a credible impact on safety, in that, the operability of the reactor missile shield support structure was not adequately justified. Subsequently, the licensee provided a basis for past operability that relied on a coefficient of friction. Therefore, because the missile shield was considered operable, this finding was of very low risk significance (Green) as determined by the Reactor Safety Significance Determination Process (Section 4OA3.6).

- Green. A Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” was identified for errors made in the calculation supporting the modification to resolve the discrepant missile shield support structure.

  This finding had the potential to affect the barrier integrity and initiating events cornerstones and was greater than minor because it had a credible impact on safety, in that, the modification as first proposed, did not adequately restore the design basis. Because the missile shield was considered operable with this inadequate design change, this finding is of very low risk significance (Green) as determined by the Reactor Safety Significance Determination Process (Section 4OA3.6).

- Green. A Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” was identified for failure to apply a design basis load in the initial modification to replace the control rod drive mechanism housing and in the calculation to determine the critical crack size for the housings.

  This finding had the potential to affect the barrier integrity and initiating events cornerstones and was greater than minor because it had a credible impact on safety, in that, the modified housing may not have performed its safety function during a seismic event and the calculated critical crack size was not conservative. Subsequently, all of the CRDM housings were replaced with newly fabricated housings, instead of installing the modified housing. Therefore, this finding did not result in an actual degradation of the primary coolant boundary and is of very low risk significance (Green) as determined by the Reactor Safety Significance Determination Process (Section 4OA3.6).

- Green. A Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” was identified for failure to consider flow affects on control rod function in the calculation of critical crack size for the housings.

  This finding had the potential to affect the barrier integrity and initiating events cornerstones and was greater than minor because it had a credible impact on safety, in that, the operability of a control rod was not justifiable given the leak rate from a postulated crack associated with a weld overlay repair for the CRDM housing. Subsequently, the licensee revised the calculation and concluded that control rod function would not have been affected by the postulated leak Therefore, this finding was determined to be of very low risk significance (Green) by the Reactor Safety Significance Determination Process (Section 4OA3.6).

Report Details

Summary of Plant Event

Beginning on June 9, 2001, the licensee 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 the licensee identified an active steam/primary coolant system leak from control rod drive mechanism (CRDM) 21. The plant was then placed in Cold Shutdown (Mode 5) and the licensee assembled a project team to evaluate the occurrence, assess extent of condition, determine root cause, and develop strategies for restoration. Initial examinations revealed that an axially-orientated flaw had propagated from the inside surface of the eccentric reducer to pipe weld (referred to as Weld 3 in this report) on the housing of CRDM 21. Subsequent examinations, identified indications of cracks on multiple housings at the Weld 3 location caused by transgranular stress corrosion cracking (TGSCC). To correct this condition, the licensee implemented replacement of all 45 CRDM 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 licensee performance, and to the extent practicable, independently validate licensee conclusions in areas including root cause determination, adequacy of repair, and corrective actions. A Special Inspection Charter was developed to focus the review 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. This inspection was the second of two efforts initiated to accomplish review of the nine charter areas. The first inspection (report 50-255/01-11) was completed on August 9, 2001, and sufficient inspections were completed in four of the nine Charter areas. However, Charter Items 2, 4, 5, 6 and 7 were not ready for inspection and remained open. Based on this inspection the Special Inspection Charter is considered complete.

Summary of Plant Status:

The Palisades plant remained in cold shutdown for CRDM replacement activities throughout the on-site inspection period.

4. OTHER ACTIVITIES [OA]

4OA3 Event Followup (93812)

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

a. Inspection Scope

The inspectors reviewed the licensee's root cause investigation report, metallographic examination report and interviewed root cause team members to assess the effectiveness of the corrective actions at preventing recurrence of CRDM housing cracks.

The requirements of 10 CFR Part 50, Appendix B, Criterion XVI "Corrective Action" were used by the inspectors to determine the acceptability of these activities.

b. Findings

The licensee identified circumferential and axial cracking in the eccentric reducer butt-weld location (termed "Weld 3") in the type 347 stainless steel (SS) housings for 29 of the 45 housings. The licensee's root cause assessment team determined that these cracks were caused by TGSCC. The contributing causes were rough machining and grinding marks on the inner bore near Weld 3, relatively stagnant air/water environment and the number of outages which have imposed pressure cycles that contribute to propagation of the cracks. The investigation and root cause determination for the cracking was documented in an attachment to Condition Report CPAL 0102186. The inspectors considered that the scope of the root cause investigation was comprehensive and that the method used was systematic.

The root cause investigation was supported primarily by the metallurgical analysis performed on the removed section of CRDM 21 housing at Weld 3. This analysis dated August 21, 2001, provided a detailed characterization of 4 cracks associated with Weld 3 on CRDM 21 and concluded that these cracks were caused by TGSCC. In this report, the through-wall axial flaw which caused the leakage event, was measured and found to be 2.8 inches in length with a 4 to 1 aspect ratio. The next most significant crack was a 2.6 inch long circumferentially oriented flaw in the counterbore radius just below Weld 3 with a measured 93 percent through-wall extent. The licensee had used visible crack growth rings to determine average crack growth rates, which were found to be 8.8×10^{-6} inches per hour and 9.6×10^{-6} inches per hour for the axial and circumferential cracks respectively. However, the inspectors noted that the licensee did not use available information to determine a crack incubation period (e.g., time required before the cracks began to grow). The inspectors considered that this could be useful information in determining how long other susceptible plant components should remain in service without examination. Based on the observed crack growth rate, the inspectors estimated that the incubation time was approximately 20 years for the cracks found in Weld 3 on the CRDM 21 housing.

The corrective actions documented in CPAL 0102186 included development and implementation of an inspection, repair and replacement program for the CRDM housings. At the conclusion of this inspection all 45 housings had been replaced with newly fabricated housings. These new housings incorporated design improvements discussed in Section 4OA3.5 that should preclude recurrence of TGSCC in the housing welds for the remainder of the plant's operating license.

The actions documented in CPAL 0102186 to prevent recurrence of TGSCC included development and implementation of an inspection plan for components potentially susceptible to TGSCC. A table of potentially susceptible components was developed with a basis documented for excluding components from additional nondestructive examinations. This table was reviewed and evaluated by the inspectors as discussed in Section 4OA3.4.

.4 <u>Extent of Condition Evaluation</u>

a. <u>Inspection Scope</u>

The inspectors reviewed records of corrective actions implemented for past cracks found in the CRDM seal housings, to determine if the extent of condition reviews/actions should have identified the Weld 3 housing cracks prior to the CRDM 21 leakage event. This included review of the corrective actions documented in the licensee event report (LER) 50-255/2001-002-00 Control Rod Drive Seal Housing Leak and Crack Indications. The inspectors also reviewed the root cause investigation report to assess the licensee's extent of condition evaluation review of other plant components potentially susceptible to TGSCC.

The requirements of 10 CFR Part 50, Appendix B, Criterion XVI "Corrective Action" were used by the inspectors to determine the acceptability of these activities.

b. <u>Findings</u>

b.1 <u>Past Corrective Actions for CRDM Seal Housing Cracks</u>

The NRC had previously identified Unresolved Item (URI) 50-255/01-11-02 associated with evaluating the adequacy of the licensee's historical corrective actions for past seal housing cracking. Specifically, this review was performed to determine if past corrective actions should have prompted identification of cracks at the Weld 3 location prior to the CRDM 21 leakage event.

The CRDM seal housings attach to the upper flange of the eight inch diameter CRDM housing. Over the life of the plant, the licensee had identified a total of six CRDM seal housings with through-wall cracks. The cause of this cracking was TGSCC and was observed in both type 304 and type 347 SS seal housings. The licensee had believed that the environmental factors and residual stress levels in these seal housings, made them uniquely susceptible locations to TGSCC. The key differences that influenced the licensee's decision to not expand the scope to lower housing welds (e.g. Weld 3) following identification of cracking at the upper seal housing included the following:

- Seal housing cracking was attributed to unfavorable (high) residual tensile stress imposed on the inside surface of the housing and tool access tube created by the vendor heat treating process or weld repairs. The Weld 3 housing location did not receive a post weld heat treatment and had not been weld repaired.

- Seal housing cracks occurred at the flange-to-housing weld, which represented a substantially different geometry than the pipe-to-pipe weld configuration at Weld 3.

- The seal housing internal environment was stagnant with a higher dissolved oxygen content than that experienced at lower elevation housing welds.

- The only applicable industry data on failures in CRDM housings which existed prior to the CRDM 21 leakage event, was from Fort Calhoun. At Fort Calhoun two spare CRDM housings were found to have TGSCC cracks that initiated from the internal weld buildup area above the Weld 3 location. The failure was attributed principally to the high oxygen environment in the non-vented spare housings. The licensee believed that the oxygen environment at the Weld 3 location was lower than that experienced at Fort Calhoun. This belief was based on the fact that Palisades housings contained active control rod mechanisms, which allowed some venting of trapped oxygen through mechanical seals during plant startup and operation.

Therefore, based on these facts, the inspectors concluded that the licensee's actions for extent of condition review based on past CRDM seal housing cracks were reasonable and could not have been expected to prevent the CRDM 21 leakage event. URI 50-255/01-11-02 is considered closed.

The nondestructive examination history of the CRDM housing welds was evaluated by the inspectors for compliance with the American Society of Mechanical Engineers (ASME) Code requirements. The ASME Code, Section XI, Table IWB-2500, Category B-O "Pressure Retaining Welds in Control Rod Housings" allowed surface or volumetric examinations of peripheral housing welds. The licensee had met the Code requirements during past outages by performing surface examinations of the peripheral CRDM housing welds. However, these surface examinations would not have been capable of detecting the TGSCC observed at the weld 3 locations, which initiated at the inside surface and progressed through the weldment. Therefore, the licensee had met ASME Code requirements, and these requirements could not have been expected to prevent the CRDM 21 leakage event.

b.2 <u>CRDM 22 Seal Housing Leak</u>

The inspectors identified a Non-Cited Violation (Green) for operation of the plant with pressure boundary leakage from a through-wall crack in the CRDM 22 seal housing.

<u>Description:</u>

(Closed) LER 50-255/01-002-00: Control Rod Drive Seal Housing Leak and Crack Indications. In this LER, cracks were identified in CRDM 22 and CRDM 8 seal housings.

The inspectors reviewed the licensee corrective actions for this condition as documented in CPAL 0101017. Based on the review of corrective actions discussed below, this LER is closed.

On March 31, 2001, the licensee identified boric acid deposits and water at the CRDM 22 seal housing location. Subsequent investigation identified a 0.7 inch long circumferential through-wall crack. In addition, the licensee identified a 0.15 inch long indication in the CRDM 8 seal housing which was not through-wall. The cause of the through-wall crack in the CRDM 22 seal housing was attributed to TGSCC of the Type 347 stainless steel (SS) housing material. The licensee replaced CRDM 8 and 22 and 11 other seal housings with new Inconel housings and on May 10, 2001, returned the remaining type 347 SS housings to service.

The licensee had found type 347 housings with through wall cracks caused by TGSCC in each of the two previous outages beginning in December of 1998. In 1999, two CRDM seal housings were identified with through-wall cracks. These, housings were replaced and the remaining housings were returned to service after non-destructive examinations. Despite these actions, two CRDM seal housings (CRDM 8 and 22) were identified as having cracks after approximately 14 months of operation. This indicated that prior licensee corrective actions to nondestructively examine and return the type 347 SS housings to service was not effective at preventing repeat housing failures.

<u>Analysis</u>:

This self revealing finding affected the barrier integrity and initiating events cornerstones. This finding was greater than minor because, if left uncorrected, it could have resulted in further degradation of the reactor coolant pressure boundary. Based on insights from fracture mechanics and leak-before-break perspectives, it was judged that operation with this degraded CRDM seal housing would not substantially increase initiating event frequency. Therefore the risk significance was very low.

<u>Enforcement</u>:

Technical Specification (TS) 3.4.13 requires that primary coolant operational leakage shall be limited to “No pressure boundary LEAKAGE” when in Modes 1 through 4. The TS 3.4.13 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 that the pressure boundary leakage from CRDM 22 housing began could not be precisely determined, it is clear that the leakage existed for greater than the 6 hour time limit to place the plant in Mode 3 required by TS 3.4.13. Therefore, contrary to the above, during the previous operating cycle which ended on March 30, 2001, the plant was not placed in Mode 3 with the pressure boundary leakage from the CRD 22 housing. This finding is considered a violation of TS 3.4.13. Because of the very low safety significance, this violation is being treated as a Non-Cited Violation (NCV 50-255/01-015-01) consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is documented in the corrective action program in CPAL 0101017.

b.3 Inadequate Preventative Actions for Type 347 Seal Housing Cracks

The inspectors identified a Non-Cited Violation (Green) for failure to implement corrective actions to prevent recurrence of cracking identified in the Type 347 SS CRDM seal housings.

Description:

On December 12, 2001, the inspectors identified that licensee corrective actions documented on CPAL 0101017, for CRDM 22 seal housing cracking were not adequate to prevent a potential failure of the remaining type 347 SS CRDM seal housings.

The licensee had found type 347 housings with through-wall cracks caused by TGSCC in each of the two previous outages beginning in December of 1998. In 1999, two CRDM seal housings were identified with through-wall cracks. These housings were replaced and the remaining housings were returned to service after non-destructive examinations. Despite these actions, two CRDM seal housings (CRDM 8 and 22) were identified as having cracks after approximately 14 months of operation. The licensee replaced these and 11 other seal housings with new Inconel housings and returned the remaining type 347 SS housings to service following the refueling outage on May 10, 2001. The service life of some type 347 SS housings returned to service apparently was less than one operating cycle, based on the previous plant history of CRDM seal failures. Specifically, using crack growth rates which could be derived from the licensee's previous seal housing failures, the type 347 SS housings may not have a sufficient service life to support a full 18 month operating cycle.

The potentially limited service life of the seal housings was not considered in the licensee's corrective actions documented in CPAL 0101017. The licensee corrective actions included implementation of a plan to replace the type 347 SS housings with Inconel housings, and had a required completion date of December 15, 2002, which was after the next scheduled outage in October of 2002. Thus, if licensee replacement plan actions were implemented at the scheduled due date, the existing type 347 housings would have remained in service for more than one operating cycle. The inspectors' questions on the limited service life appeared to prompt the licensee decision to replace the remaining type 347 seal housings. On December 10, 2001, the inspectors were informed by the licensee that the remaining type 347 SS seal housings would be replaced prior to plant restart.

Analysis:

This finding had the potential to affect the barrier integrity and initiating events cornerstones. This finding was greater than minor because, if left uncorrected, it could have resulted in breach of the reactor coolant pressure boundary at the type 347 SS CRDM seal housings. Fortuitously, the licensee operated for only one month after returning these housings to service. Therefore, no actual degradation of the primary pressure boundary occurred and the risk significance was very low (Green) as determined by the Reactor Safety Significance Determination Process determination process.

Enforcement:

10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires in part that for significant conditions adverse to quality that corrective actions are taken to prevent recurrence. In this case, the identified defective material (Type 347 SS CRD seal housings) had repetitively experienced cracking, which is a significant condition adverse to quality. Contrary to the above, the licensee failed to implement actions to prevent recurrence (repetitive failure) of the type 347 SS CRDM housings returned to service on May 10, 2001. Specifically, the service life of the housings was in question and no demonstration of service life had been documented which supported the corrective action (replacement) due date of December 15, 2002. This finding is considered a violation of 10 CFR Part 50, Appendix B, Criterion XVI. Because of the very low safety significance, this violation is being treated as a Non-Cited Violation (NCV 50-255/01-015-02) consistent with Section VI.A.1 of the NRC Enforcement Policy. This finding is documented in the corrective action program in CPAL 0104061 and CPAL 0104062.

b.4 Operation with Pressure Boundary Leakage from CRDM 21

A violation of TS 3.4.13 was identified for operation of the plant with pressure boundary leakage from a through-wall crack in the CRDM 21 housing.

The NRC had previously identified URI 50-255/01-11-01 associated with a potential violation of TS 3.4.13 due to plant operation with pressure boundary leakage from a through-wall crack in CRDM 21 at Weld 3. The licensee did not have data on the magnitude of the unidentified leakage attributed specifically to CRDM 21 at the time of plant shutdown, but the total unidentified primary coolant system leak rate was approximately 0.3 gallons per minute. In addition, the licensee identified 28 other CRDM housings with crack indications as discussed in Section 4OA3.4.b.5, which had not breached the pressure boundary.

Technical Specification 3.4.13 requires that primary coolant operational leakage shall be limited to "No pressure boundary LEAKAGE" when in Modes 1 through 4. The TS 3.4.13 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 that the pressure boundary leakage from CRDM 21 housing began could not be precisely determined, it is clear that the leakage existed for greater than the 6 hour time limit to place the plant in Mode 3. Therefore, contrary to the above, during the previous operating cycle which ended on June 20, 2001, the plant was not placed in Mode 3 with the pressure boundary leakage from the CRD 21 housing. However, as discussed in Section 4OA3.4.b.1, the inspectors concluded that existing Code requirements and corrective actions for past seal housing cracks could not have been expected to prevent the CRDM 21 leakage event. Therefore, since this TS violation resulted from equipment failure not avoidable by reasonable quality assurance measures or management controls, discretion is exercised in accordance with section VII.B.6 of the NRC Enforcement Policy and a notice of violation will not be issued. URI 50-255/01-11-01 is considered closed.

The inspector considered the risk of past operation with multiple degraded CRDM housings with respect to the increased likelihood of a loss of coolant accident. Based on the critical flaw size relative to the existing flaw sizes, substantial structural margins existed with respect to catastrophic housing failure. Further, the calculated leakage at one half the critical crack size was sufficient to require a plant shutdown; thus, the licensee would have experienced leak indications and shut down well in advance of developing a critical flaw thus precluding large and medium break loss of coolant accidents. The NRC risk analyst determined that the initiating event frequency would have to increase by a factor of 10 before the issue would reach low risk significance. Given that substantial structural margin existed, the increase in the initiating event frequency was considered to be much less than a factor of 10. Therefore, this finding is considered of very low risk significance.

b.5 Multiple degraded CRDM Housings Identified During Extent of Condition Review

The licensee performed external volumetric examinations and internal visual examinations of the original CRDM housings as documented in report 50-255/01-11. Based on this examination, the licensee identified circumferential and axial cracking near the Weld 3 housing location in 28 of the 45 housings (excluding the failed CRDM 21 housing). The largest circumferential crack indication was 4.2 inches in length and the largest axial crack indication was 2.25 inches in length and 14 of these housings had multiple rejectable crack indications in the same housings.

To correct the degraded housings, the licensee implemented housing replacement as discussed in Section 4OA3.5. The CRDM housing replacement scope did not include Weld 1 the reactor head nozzle to CRDM flange weld. For this weld location, the licensee had performed ultrasonic examinations on 24 CRDM housing locations accessible from the periphery of the vessel head. By letter dated November 6, 2001, the licensee submitted to the NRC, their technical basis for concluding that this weldment had a low susceptibility to TGSCC. The primary basis for this conclusion was due to the lower welding residual stresses and smooth surface finish at the inside diameter of Weld 1. The inspectors reviewed the licensee's calculation of welding residual stresses and had no further concern for this weldment.

To evaluate the potential for undetected cracking in other austenitic stainless steel plant components, the licensee developed a table of components potentially susceptible to TGSCC. The licensee documented their basis in CPAL 0102186 for excluding many of these components from additional evaluation/examination. The inspectors questioned the licensee's basis for excluding small bore primary system piping components, which were excluded because operating stresses were expected to be low. The inspectors did not consider this factor alone to be a complete basis for excluding these components, because the welding residual stress could be sufficient to support TGSCC. The licensee subsequently initiated CPAL 0103734 and provided an additional basis to exclude these components associated with the operating environment.

From the screening of components potentially susceptible to TGSCC documented in the CPAL 0102186 table, only the incore instrument nozzle welds and CRDM housing welds were not excluded. For these components, the licensee intended to perform augmented surface and volumetric examinations during future outages, beginning with the 2003 refueling outage.

.5 Adequacy of Repair Methodology

i. Inspection Scope

On July 30, 2001, the licensee submitted a request to the NRC to use Code Case 504-1 as an option to weld overlay repair cracked CRDM housings. However, on November 19, 2001, the Site Vice President informed inspectors, that the decision had been made to replace all 45 housings with new housings. Therefore, inspectors reviewed the design change documentation for the replacement housings. The inspectors also observed the licensee performing welding of the omega seal during installation of replacement housings, observed on-site receipt inspection of two new replacement housings and reviewed radiographs for the replacement housing welds.

The requirements of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," and the American Society of Mechanical Engineers (ASME) Code, Section III, V, and XI were used by the inspectors to determine the acceptability of these activities.

j. Findings

b.1 Receipt Inspection and Welding of Replacement CRDM Housings

The majority of the receipt inspection of the replacement CRDM housings had been conducted by the licensee at the vendor's facility. The on-site checks of housings consisted of verification of housing serial numbers, supporting attachment hardware and a cursory visual check for shipping damage. The inspectors observed the on-site receipt inspection activities and considered licensee controls adequate. Inspectors noted that the housings arrived with a Code NPT stamp certification for an ASME Section III, Class 1 vessel and had appropriate shipping covers to prevent intrusion of foreign materials.

The radiographic weld records on four replacement housings were reviewed by the inspectors. These radiographs met the ASME Code1989 Edition, Section V, requirements for quality. No relevant indications were identified and the licensee had accepted these housings for installation. The inspectors also observed omega seal welding during installation of four replacement housings (CRDM 4, 11, 24 and 27). This welding was conducted in accordance with the weld procedure and no problems were identified.

b.2 Design Change for Replacement CRDM Housings

The licensee had attributed the TGSCC at Weld 3 in part, to the rough grinding marks left on the inner bore surfaces near Weld 3. To mitigate the potential for TGSCC in the new housings, a relatively smooth 125 RMS finish on the inner bore was specified. Additionally, minor changes in the housing weld locations relative to the outside diameter geometry were made such that more complete volumetric examinations of the housing welds could be performed. The licensee implemented design change EAR-

2001-0426-01, “CRD Upper Housing Redesign.” The changes from the original design included:

- a redesigned eccentric reducer machined from bar stock that eliminated Weld 2 in the original housing;
- relocation of Weld 3 to a position further away from the eccentric reducer to eliminate geometrically induced stresses which also facilitated access for volumetric examinations, and;
- application of heat sink welding to reduce weld joint residual stress, thereby reducing susceptibility to TGSCC.

These design changes adequately resolved the root/contributing causes of the CRDM 21 leak event and should prevent recurrence of housing failure by TGSCC for the remainder of the plant operating license.

b.3 Inadequate Crack Growth Rate Used in Proposed for Overlay Repair

The inspectors identified a noncited violation (Green) for use of a nonconservative crack growth rate in a calculation supporting a weld overlay repair design change for the CRDM housings.

Description:

Previously the NRC identified URI 50-255/01-11-03, associated with a concern for the adequacy of a crack growth rate used to support the original weld overlay repair design. The supporting calculation for this repair was documented in EA-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.” The inspectors questioned the adequacy of the original overlay design which evaluated the acceptability of the weld overlay repair based on a crack growth rate derived mainly from data associated with intergranular stress corrosion cracking. The inspectors calculated a crack growth rate of approximately three times the licensee’s rate of 4.5×10^{-6} inches per hour, based on TGSCC data from two spare CRDM housings at Fort Calhoun. The licensee previously determined in CPAL 9902295, that the Fort Calhoun crack growth rate was consistent with crack growth rates observed in failed Palisades CRDM seal housings (also type 347 SS).

The use of inappropriate crack growth rates could have potentially resulted in installation of a weld overlay repair that was of insufficient length (due to higher crack growth rates) to ensure the integrity of the primary coolant pressure boundary. The licensee subsequently used a bounding crack growth rate value of 2×10^{-5} inches per hour in the request to use Code Case N-504-1 submitted to the NRC on July 30, 2001, to justify a full structural weld overlay repair design.

Analysis:

This finding had the potential to affect the barrier integrity and initiating events cornerstones. This finding was greater than minor because, if left uncorrected, it could have resulted in installation of an inadequate overlay repair for a degraded the primary coolant pressure boundary. The inspectors noted that no process checks existed which would have identified the inadequate crack growth rate prior to implementing the overlay modification. Subsequently, the control rod drive housings were replaced and the weld overlay design change was not implemented. Therefore, the integrity of the primary coolant system boundary was not affected and this finding was determined to be of very low risk significance (Green) by the Reactor Safety Significance Determination Process.

Enforcement:

10 CFR Part 50, Appendix B, Criterion III "Design Control" states in part that measures shall provide for checking the adequacy of design. Contrary to the above, the licensee failed to adequately check the suitability of the weld overlay design documented in EA-EAR-2001-0373-01, in that a nonconservative crack growth rate was used to confirm/check the adequacy of the design. This finding is considered a violation of 10 CFR Part 50, Appendix B, Criterion III. Because of the very low safety significance, this violation is being treated as a Non-Cited Violation (NCV 50-255/01-15-03) consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is documented in the corrective action program in CPAL 0103150. URI 50-255/01-11-03 is considered closed.

.6 Adequacy of Overall Corrective Actions

a. Inspection Scope

The inspectors reviewed the initial corrective actions and interviewed members of the root cause team to assess the adequacy of the corrective actions. Included in this review were corrective actions documented in the licensee event report (LER) 50-255/2001-004-00 Control Rod Drive Mechanism Upper Housing Assembly Crack Indications.

The NRC had previously identified a number of Unresolved Items associated with engineering issues documented in Special Inspection Report 50-255/01-11. Corrective actions for these issues were reviewed by the inspectors during this inspection.

The requirements of 10 CFR Part 50, Appendix B, Criterion III "Design Control" and Criterion XVI "Corrective Action" were used by the inspectors to determine the acceptability of these activities.

b. Findings

b.1 Initial Corrective Actions

(Closed) LER 50-255/2001-004-00 Control Rod Drive Mechanism Upper Housing Assembly Crack Indications. The licensee documented identification of a through-wall crack in CRDM-21 and cracking at 28 other CRDM upper housings at the Weld 3 location. The licensee had formed a root cause team to investigate the cause of the cracking and had removed the CRDM-21 housing to perform non-destructive and destructive examinations which characterized the cracking and confirmed the root cause. The root cause team investigation and proposed corrective actions were reviewed and considered adequate by the inspectors as discussed herein. Because the leak was small and all accident mitigation equipment was available, the CRDM leakage event was not considered risk significant as documented in NRC report 50-255/01-11. Based on the review of corrective actions, this LER is closed.

b.2 Engineering Design Control Errors

b.2.1 Unanalyzed Missile Shield Support Structure

b.2.1.1 Inadequate Operability Evaluation for Reactor Missile Shield Support Structure

The inspectors identified a Non-Cited Violation (Green) for failure to consider bending loads at the I-beam's web in the initial operability evaluation of the discrepant missile shield support structure.

Description:

During the investigation of CRDM housing cracks, the licensee identified a discrepant configuration on the missile shield support structure over the reactor. The licensee evaluated this as built configuration in the operability evaluation for CPAL 0102248. The evaluation concluded that the different configuration of the missile shield support structure had no effect on the seismic requirements.

The proposed corrective action for this issue was to update the affected structural drawings to reflect the as-built configuration of the missile shield support structure. With the lower I-beam flange bolted to the floor and lateral seismic loads applied near the upper flange from the 250,000 pound missile shield, the weak-axis bending of beam's web was critical to the structure's stability. However, the inspectors identified that the bending stress in the web of the 36-inch I-beam support structure had not been evaluated. The inspectors estimated that the laterally-loaded 36-inch I-beam in the missile shield support structure would be significantly overstressed during a design basis seismic event. Therefore, had the NRC not identified this issue, during a seismic event, the overstressed support beam may have failed, allowing the missile shield to fall onto the reactor vessel head. The impact of the missile shield would likely have caused a loss-of-coolant-accident and damage to control rods, which could have prevented a reactor shut down.

NRC questions, prompted the licensee to perform a revised operability evaluation for the missile shield support structure. The licensee concluded that the structure would have remained operable during a seismic event. The missile shield support configuration was subsequently modified to restore the design basis.

Analysis:

This finding had the potential to affect the barrier integrity and mitigating systems cornerstones. This finding was greater than minor because it had a credible impact on safety, in that, the initial operability determination of the reactor missile shield support structure was not adequately demonstrated. Subsequently, the licensee provided a basis for past operability that relied on engineering judgement associated with the coefficient of friction between the missile shield and the support structure. Therefore, because the missile shield was considered operable, this finding was of very low risk significance (Green) as determined by the Reactor Safety Significance Determination Process.

Enforcement:

10 CFR Part 50, Appendix B, Criterion III "Design Control" requires, in part, that measures shall provide for checking the adequacy of design. Contrary to the above, the licensee failed to check the adequacy of the missile shield support structure in operability evaluation CPAL 0102248, in that, a critically stressed component (e.g. bending stress in the web of the 36-inch I-beam) was not evaluated. This finding is considered a violation of 10 CFR Part 50, Appendix B, Criterion III. Because of the very low safety significance, this violation is being treated as a Non-Cited Violation (NCV 50-255/01-15-04) consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is documented in the corrective action program in CPAL 0102647. URI 50-255/01-11-04 is considered closed.

b.2.1.2 Inadequate Modification of Reactor Missile Shield Support Structure

The inspectors identified a Non-Cited Violation (Green) for errors made in the calculation supporting the modification to resolve the discrepant missile shield support structure.

Description:

The licensee proposed a modification of the missile shield support structure to restore the design basis. To resolve the discrepant condition, the proposed modification included stitch welds as documented in calculation EA-EAR-2001-0385-01, "Evaluation of Missile Shield Support." The inspectors identified errors in evaluating components in the load path to the building structure for this calculation as discussed below:

- the effect of prying on 5x5x1/4 curb angles, strap anchors and Nelson stud anchors had not been considered;
- the calculated vertical force on curb angle embedment straps did not include load amplification affects from the stitch weld spacing;
- the curb angle stress evaluation used a non-conservative flange effective width;
- the impact of the bolted connection between the 12-inch channels was not considered because the licensee had no information regarding the bolts sizes or spacing;
- drawing 8-C-147 Sheet 1 Revision A, for this modification identified a fillet weld installation. This configuration was physically impossible to install because of the existing field misalignments at one end of the channel to angle interface.

After reviewing these issues, the licensee concluded that the capacity of the restraints of the curb angles was "somewhat limited" and "due to a number of factors," the structure's attachment to the curb would be conservatively neglected. Based on this conclusion, the licensee initiated a modification to add bracing members to the sides of the support structure which restored compliance with the design basis. This final design change was evaluated in calculation EA-EAR-2001-0591-03, "Provide Alternate Load Path for Reactor Vessel Missile Shield Supports." No further issues were identified with this calculation.

Analysis:

This finding had the potential to affect the barrier integrity and mitigating systems cornerstones. This finding was greater than minor because it had a credible impact on safety, in that, the modification as first proposed in calculation EA-EAR-2001-0385-01, did not adequately restore the design basis as intended. This calculation was not considered work-in-progress because no further process checks existed which would have identified these errors. Because the missile shield was still considered operable with this inadequate design change, this finding is of very low risk significance (Green) as determined by the Reactor Safety Significance Determination Process

Enforcement:

10 CFR Part 50, Appendix B, Criterion III "Design Control" states, in part, that measures shall provide for checking the adequacy of design. Contrary to the above, the licensee failed to check the adequacy of the missile shield support structure in calculation EA-EAR-2001-0385-01, "Evaluation of Missile Shield Support," in that, critical components (e.g. curb angles, strap anchors and Nelson stud anchors) were not adequately evaluated. This finding is considered a violation of 10 CFR Part 50, Appendix B, Criterion III. Because of the very low safety significance, this violation is being treated as a Non-Cited Violation (NCV 50-255/01-15-05) consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is documented in the corrective action program in CPAL 0103880.

b.2.2 Design Basis Loading for CRDM Housings

The inspectors identified a Non-Cited Violation (Green) for failure to apply a design basis load in the initial modification to replace the CRDM housing and in the calculation to determine the critical crack size for the housings.

Description:

The inspectors identified that the licensee failed to use a design basis load in determining the critical crack size for the CRDM housings and in evaluating the adequacy of the replacement housing for CRDM-21. The design basis calculation for the CRDM housing was documented in Combustion Engineering Report No. TR-ESE-437, "Palisades CRDM Dynamic Analysis Report," July 6, 1981. However, because of the lack of details in this report, the licensee chose to use information from the initial analysis of the CRDM housing completed in 1967. Although the initial analysis appeared to apply a bounding bending moment, it did not include an 18,000 pound axial force on the CRD housing as described in the 1981 design basis analysis. Consequently, a potentially non-conservative critical crack size was calculated in EA-EAR-2001-0373-04, Attachment 1, “Evaluation of Leakage from Circumferential and Axial Through-Wall Cracks in Lower CRDM Housing.” Had the NRC not identified this issue, a non-conservative critical crack size, could have been applied to future decisions on the acceptability of flaws identified in the CRDM housings.

The 18,000 pound axial load discussed above, was not also not considered in modification EAR-2001-0382, “CRD Upper Housing Replacement.” This error was significant because it allowed the licensee to incorrectly conclude that the seismic restraint collar (weld buildup area) on the original CRDM housings functioned only to facilitate the installation of the seismic restraint. Thus, the proposed modification which included replacing the original collar design with two rings attached to the housing using 1/8-inch skip welds, would not have been adequate to withstand the axial design loads.

Analysis:

This finding had the potential to affected the barrier integrity and mitigating systems cornerstones. This finding was greater than minor because it had a credible impact on safety, in that, the modified housing may not have performed its safety function during a seismic event and the calculated critical crack size was not conservative. This calculation was not considered work-in-progress because no further process checks existed which would have identified these errors. Subsequently, the licensee decided to replace the all of the CRDM housings with newly fabricated housings, instead of installing the modified housing. Therefore, this finding did not result in an actual degradation of the primary coolant boundary and is of very low risk significance (Green) as determined by the Reactor Safety Significance Determination Process.

Enforcement:

10 CFR Part 50, Appendix B, Criterion III “Design Control” requires, in part, that measures shall provide for checking the adequacy of design. Contrary to the above, the licensee failed to check the adequacy of the modified control rod drive housing and the calculated critical crack size, in that, significant seismic loads were not considered in either case. This finding is considered a violation of 10 CFR Part 50, Appendix B, Criterion III. Because of the very low safety significance, this violation is being treated as a Non-Cited Violation (NCV 50-255/01-15-06) consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is documented in the corrective action program in CPAL-0103817. URI 50-255/01-11-05 is considered closed.

b.2.3 Leak Rate Flow Effect on Rod Function

The inspectors identified a Non-Cited Violation (Green) for failure to consider flow affects on control rod function in the calculation of critical crack size for the housings.

Description:

The inspectors identified that the licensee failed to consider the effect of leakage flow on the function of the control rod in a calculation supporting a proposed weld overlay repair for the cracked housings. The licensee performed a calculation of the critical crack size for the CRDM housing as documented in calculation EA-EAR-2001-0373-01, Attachment 4, “Safety Assessment Report for the Palisades Nuclear Plant Control Rod Drive Mechanism Weld Overlay.” However, this calculation considered only the structural stability of the housing with a postulated crack and neglected the impact of upward flow through the CRDM nozzle on the function of the control rod. The licensee failed 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 inserting as a result of the high differential pressure between the reactor and the housing due to the flow restrictions through the CRDM nozzle. The inspectors’ questions prompted the licensee to perform a revised calculation which adequately resolved this issue.

Analysis:

This finding had the potential to affect the barrier integrity and mitigating systems cornerstones. This finding was greater than minor because it had a credible impact on safety, in that, the operability of a control rod was not justifiable given the leak rate from a postulated crack associated with a weld overlay repair for the CRDM housing. This calculation was not considered work-in-progress because no further process checks existed which would have identified this error. Subsequently, the licensee revised the calculation and concluded that control rod function would not be affected by the postulated leak rate. Therefore, this finding was of very low risk significance (Green) as determined by the Reactor Safety Significance Determination Process.

Enforcement:

10 CFR Part 50. Appendix B, Criterion III, “Design Control.” states, in part, that measures shall provide for checking the adequacy of design. Contrary to the above, the licensee failed to check the adequacy of the in calculation, in that, the impact of pressure drops due to fluid flow on control rod functions were not considered. This finding is considered a violation of 10 CFR Part 50, Appendix B, Criterion III. Because of the very low safety significance, this violation is being treated as a Non-Cited Violation (NCV 50-255/01-15-07) consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is documented in the corrective action program in CPAL-0103864. Unresolved Item 50-255/01-11-06 is considered closed.

b.2.4 Rod Ejection Effect on Adjacent Rods Due to Seismic Restraint

Previously, the NRC had identified a concern (URI 50-255/01-11-07) associated with the affects on adjacent control rods from a rod ejection due to transfer of forces at the seismic restraints. Based on review of calculation W-CPC-13Q-313, “Displacement Evaluation of CRDM Housings” this issue was adequately evaluated and URI 50-255/01-11-07 is considered closed.

.7 Similarity with Other Leakage Issues

a. Inspection Scope

The inspectors reviewed the root cause investigation report attached to CPAL 0102186 to assess the licensee consideration of industry leakage events. The requirements of

10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” were used by the inspectors to determine the acceptability of this activity.

b. Findings

The licensee had used industry operating experience and previous station operating history in determining the root cause and extent of condition evaluations for the CRDM housing cracks. Specifically, the licensee considered; their past experience with cracking in the upper CRDM seal housings in type 304 and 347 SS, the experience French at Electricite de France cracking in type 304, 316 SS housings and, cracking in spare CRDM housings at Fort Calhoun in type 348 SS. Based on this information, the licensee concluded that ineffective post weld heat treatment and residual stresses are the dominant variables resulting in poorer housing material performance. The inspectors considered that the licensee had appropriately considered and evaluated the limited industry data on similar cracking and leakage events.

4OA4 Cross-cutting Issues

A significant cross-cutting human performance finding was identified associated with multiple examples of inadequate engineering products that provided a technical bases for modifications, operability and corrective action evaluations. This performance deficiency reflected a lack of rigor applied to preparation, checking, and verifying mechanical, structural, and metallurgical engineering products that affected the reactor safety cornerstones for initiating event frequency, barrier integrity, and mitigating systems. The NRC identified seven findings (NCVs), where the engineering products were not adequate.

(1) The licensee operated for greater than with a through-wall crack in the CRDM 22 seal housing. The licensee had an extensive history of seal housing cracking caused by TGSCC of the type 347 SS housings and corrective actions taken had not prevented recurrence of this cracking (NCV 50-255/01-15-01). This issue is discussed in 40A3.4.b.2 above.

(2) The remaining service life of the type 347 seal housings susceptible to TGSCC was not evaluated prior to returning them to service (NCV 50-255/01-15-02). This issue is discussed in 40A3.4.b.3 above.

(3) The design calculation for the proposed overlay repair used an unjustifiably low crack growth rate (NCV 50-255/01-15-03). This issue is discussed in 40A3.5.b.3 above.

(4) The initial operability evaluation of the discrepant missile shield support structure over the reactor did not consider bending loads in the I-beam’s web, which was critical to the structure’s stability (NCV 50-255/01-15-04). This issue is discussed in 40A3.6.b.2.1.1 above.

(5) The design calculation for the initial modification to fix the reactor missile shield support structure did not adequately evaluate components in the load path

to the building structure (NCV 50-255/01-15-05). This issue is discussed in 40A3.6.b.2.1.2 above.

(6) The licensee failed to apply a design basis load in the initial modification to replace the CRDM housing and in the calculation to determine the critical crack size for the housings (NCV 50-255/01-15-06). This issue is discussed in 40A3.6.b.2.2 above.

(7) The effects of leakage flow rates on control rod function were initially not considered in a calculation evaluating the critical crack size (NCV 50-255/01-15-07). This issue is discussed in 40A3.6.b.2.3 above.

The causal relationship for the above finding, was a lack of rigor applied to preparation, checking and verifying of engineering products, during the design control review process (FIN 50-255/01-15-08). Without NRC intervention, inadequate modifications may have been installed and/or degraded equipment may have been returned to service without an adequate basis to confirm operability. At the conclusion of this inspection, the licensee was in the process of performing a comprehensive review to identify and correct the causes of this adverse trend in human performance. This action appeared important to ensuring the integrity of the plant design basis.

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 January 29, 2002. The licensee acknowledged the findings presented. Proprietary information was received and reviewed by the inspectors and subsequently returned to the licensee.

KEY POINTS OF CONTACT

Licensee

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D. Cooper, Site Vice President
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A. Vegel, Chief, Branch 6, DRP

LIST OF ITEMS OPENED AND CLOSED

Opened

50-255/01-15-01	NCV	Violation of TS 3.4.13 for operation with CRDM 22 seal housing leakage (Section 4OA3.4)
50-255/01-15-02	NCV	Inadequate preventative actions for type 347 stainless steel seal housing cracks (Section 4OA3.4)
50-255/01-15-03	NCV	Failure to adequately check the suitability of the weld overlay design associated with use of a nonconservative crack growth rate (Section 4OA3.5)
50-255/01-15-04	NCV	Inadequate design control associated with the initial operability evaluation of the discrepant missile shield support structure (Section 4OA3.6)
50-255/01-15-05	NCV	Inadequate design control associated with the modification to resolve the discrepant missile shield support structure (Section 4OA3.6)
50-255/01-15-06	NCV	Inadequate design control associated with the initial modification to replace the CRDM housing with a spare housing from Fort Calhoun (Section 4OA3.6)

50-255/01-15-07	NCV	Inadequate design control associated with the failure to consider flow affects on rod function in the calculation of critical crack size for the CRDM housing (Section 4OA3.6)
50-255/01-15-08	FIN	Human performance deficiency. Inadequate engineering products for modifications, operability and corrective action evaluations (Section 4OA4)

<u>Closed</u>

50-255/01-15-01	NCV	Violation of TS 3.4.13 for operation with CRDM 22 seal housing leakage (Section 4OA3.4)
50-255/01-15-02	NCV	Inadequate preventative actions for type 347 stainless steel seal housing cracks (Section 4OA3.4)
50-255/01-15-03	NCV	Failure to adequately check the suitability of the weld overlay design associated with use of a nonconservative crack growth rate (Section 4OA3.5)
50-255/01-15-04	NCV	Inadequate design control associated with the initial operability evaluation of the discrepant missile shield support structure (Section 4OA3.6)
50-255/01-15-05	NCV	Inadequate design control associated with the modification to resolve the discrepant missile shield support structure (Section 4OA3.6)
50-255/01-15-06	NCV	Inadequate design control associated with the initial modification to replace the CRDM housing with a spare housing from Fort Calhoun (Section 4OA3.6)
50-255/01-15-07	NCV	Inadequate design control associated with the failure to consider flow affects on rod function in the calculation of critical crack size for the CRDM housing (Section 4OA3.6)
50-255/01-15-08	FIN	Human performance deficiency. Inadequate engineering products for modifications, operability and corrective action evaluations (Section 4OA4)
50-255/01-11-01	URI	NRC Review of TS 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
50-255/2001-002-00	LER	Control Rod Drive Seal Housing Leak and Crack Indications
50-255/2001-004-00	LER	Control Rod Drive Mechanism Upper Housing Assembly Crack Indications

LIST OF ACRONYMS USED

ASME	American Society of Mechanical Engineers
CRDM	Control Rod Drive Mechanism
LER	Licensee Event Report
NCV	Noncited Violation
URI	Unresolved Item
SS	Stainless Steel
TGSCC	Transgranular Stress Corrosion Cracking

LIST OF DOCUMENTS REVIEWED

Calculations

EA-CPAL-01-2186-01	Assessment of Stress Levels of CRD Upper Housing Weld Joints	Revision 0
EA-CPAL-01-2186-02	CRD Upper Housing and Nozzle Weld Susceptibility Comparison	Revision 1
A018000230281	Design Analysis of the Palisades CRDM Pressure Housing	March 7, 1968
EA-EAR-2001-0402-06	Crack Growth Rate Calculation	Revision 0
EA-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	Revision 1
EA-EAR-2001-0385-01	Evaluation of Missile Shield Support	Revision 0
EA-EAR-2001-0591-01	Evaluation of Missile Shield Support Structure	Revision 0
EA-EAR-2001-0591-02	Detailed ANSYS Computer Analysis of the Missile Shield Support Structure,	Revision 0
EA-EAR-2001-0591-03	Provide Alternate Load Transfer Path for Reactor Vessel Missile Shield Supports	Revision 0
Combustion Engineering Report No. TR-ESE-437	Palisades CRDM Dynamic Analysis Report	July 6, 1981
EA-EAR-2001-0373-04, Attachment 1	Evaluation of Leakage from Circumferential and Axial Through-Wall Cracks in Lower CRDM Housing	Revision 4
EA-EAR-2001-0373-01, Attachment 4	Safety Assessment Report for the Palisades Nuclear Plant Control Rod Drive Mechanism Weld Overlay	Revision 1
W-CPC-13Q-313	Displacement Evaluation of CRDM Housings	Revision 0

Condition Reports

CPAL-0102186	Primary Coolant System Pressure Boundary Leakage CRD-21 Upper Housing Assembly	
CPAL-992295	CRDM Housing Crack and Indication	
CPAL-0101017	Suspected PCS Boundary Leakage on Control Rod #22 Seal Housing	
CPAL-0103852	Preliminary Examination of CRD Samples Identified Conditions Requiring Further Evaluation	
CPAL-0103734	Extent of Condition Not Sufficient	
CPAL0102987	Potential Adverse Trend in Control Rod Drive Pressure Housing Replacement Project	
CPAL0103797	NRC Inspector Identified Potential Weaknesses in Engineering	
CPAL0103799	Replacement CRD Engineering Analysis Referenced Inappropriate Design Criteria	

Design Changes

EAR-2001-0426-01	CRD Upper Housing Redesign.	Revision 1
EAR-2001-0382	CRD Upper Housing Replacement	Revision 0

Drawings

CND-E-5003	Upper Housing Replacement	Revision 5
8-C-147 Sheet 1	Missile Shield Support	Revision A

Other Documents

0100427	Palisades CRD-21 Upper Housing Metallographic Examination	August 24, 2001
Letter	Request for Approval to Use ASME Code Case N-504-1 for Repair Of Control Rod Drive Mechanism Upper Housing Assemblies	July 30, 2001

Letter	Response to Request for Additional Information Regarding Repair of Control Rod Drive Mechanism Upper Housing Assemblies (TAC No. MB3001).	November 6, 2001
LER 50-255/2001-004-00	Control Rod Drive Mechanism Upper Housing Assembly Crack Indications	Revision 0
50-255/2001-002-00	Control Rod Drive Seal Housing Leak and Crack Indications	Revision 0
50-255/1999-004-01	Control Rod Drive Seal Housing Leak and Crack Indications	Revision 1

Radiographic Weld Records

CRD serial number RP 1063)	Lower Housing Weld
CRD serial number RP 1060	Lower Housing Weld
CRD serial number RP-1086	Upper and Lower Housing Welds
CRD serial number RP-1096	Upper and Lower Housing Welds

Weld Procedures Procedure Qualification Records and Filler Material Certification

1149-1	Weld Procedure Specification	Revision 3
1149-3	Weld Procedure Specification	Revision 3
10AS024	Weld Procedure Qualification	Revision 0
10AS051	Weld Procedure Qualification	Revision 0
200163577	Weld Filler Material Certification	Revision 0