FOIA/PA NO: ________2013-0250_____

GROUP C

RECORDS BEING RELEASED IN THEIR ENTIRETY

EVENT TIMELINE

TIME	EVENT
June 2011	Protective Relay Modification Including Zone G Installed on Unit 1. Offsite power to Unit 1 aligned from Unit 2
July 7, 2011	Unit 1 'A' train essential bus aligned to Unit 1
November 5, 2011	Unit 1 'B' train essential bus aligned to Unit 1
December 15, 2011	Unit 1 shutdown occurs without causing a LOOP
February 4, 2012	Unit 1 power aligned as supply to Unit 2 'A' train essential buses
February 18, 2012	Unit 1 power aligned as supply to Unit 2 'B' train essential bus
March 10, 2012	Unit 2 shutdown for refueling outage
April 4, 2012	Unit 1 is operating at 100%. Unit 2 is in MODE 5 with ND in service. Power to Unit 2 essential buses supplied from Unit 1
8:03 p.m.	1D NCP Y Phase cable faults to ground causing trip of 1D NCP Automatic Reactor Trip on 1D NC loop low flow Automatic Turbine Trip on Reactor Trip with power > P-8 1ATD supply to essential bus 1ETB trips deenergizing the bus
8:03:10	Generator Output breakers 1A and 1B open; 1B EDG automatically starts and repowers essential bus 1ETB
8:03:25	Generator frequency decrease below 57.9 Hz causing instantaneous underfreqency protective relay to isolate Unit 1 offsite power causing Uni1 LOOP and loss of power to Unit 2; Essential buses 1ETA, 2ETA, and 2ETB deenergize
8:03:35	1A, 2A, and 2B EDGs start and repower their essential buses; Overcurrent alarm on 2A EDG
8:06	2A ND pump started to restore decay heat removal
8:12	NOUE Declared
8:16	Initial notifications made
8:30	2B SFP cooling pump started
9:22	TSC activated
10:32	EOF activated
11:03	SSF EDG started
11:06	SSF EDG declared inoperable due to low output voltage

April 5, 1:29 a.m.	Offisite power restored to 1ETA essential bus
1:37	Offisite power restored to 2ETB essential bus; NOUE terminated
1:38	1A EDG shutdown
1:43	2B EDG shutdown
2:36	Offisite power restored to 2ETA essential bus
2:45	2A EDG shutdown
5:37	Offisite power restored to 2ETA essential bus
5:41	1B EDG shutdown
9:00	SSF EDG shutdown

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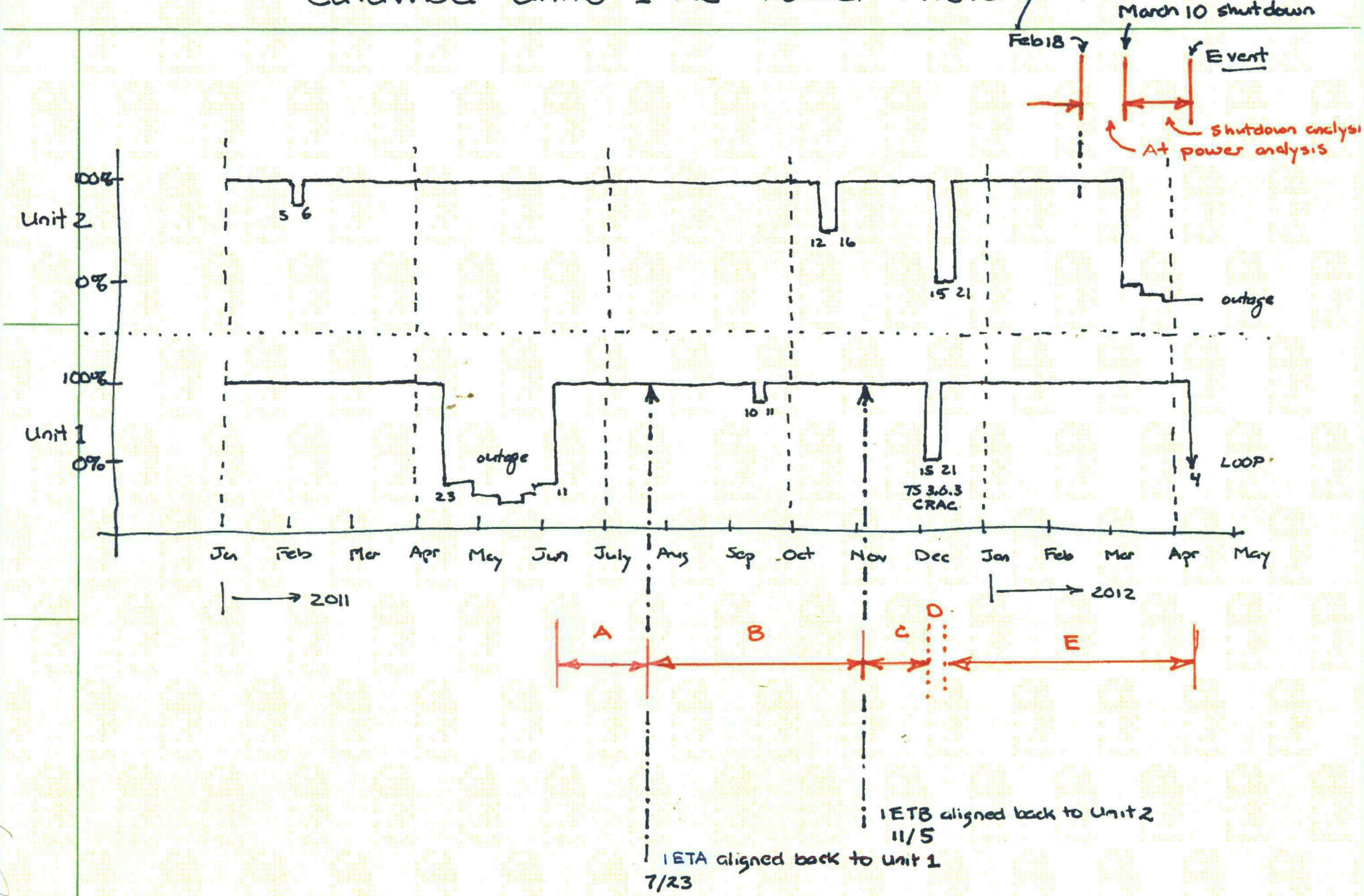
Timeline of Events - Catawba Loss of Offsite Power

Date	Time	Event/Issue/Action
7/23/11		1ETA aligned back to Unit 1 SY (was aligned to Unit 2 during outage)
11/5/11		1ETB aligned back to Unit 1 SY (was aligned to Unit 2 during outage)
2/4/12	1755	2ETA aligned to Unit 1 (1TC-4)
2/18/12	1555	2ETB aligned to Unit 1 (SATB)
3/10/12	0424	Unit 2 Turbine Offline
3/10/12	1022	Unit 2 RHR in service **(POS 1)**
3/10/12	1403	2B EDG Inoperable
3/13/12	0345	Loops not filled **(POS 2)**
3/15/12	0302	Unit 2 Water Level >23 feet **(POS 3)**
3/15/12	0315	2B ND Pump Unavailable
3/15/12	0406	2B ND Pump Available
3/16/12	0345	2B ND Pump Inoperable
3/23/12	1435	2B EDG Operable
3/23/12	1549	2A EDG Inoperable (Outage tagout)
3/27/12	0511	2B ND Pump Operable
4/1/12	0848	2A EDG Operable
4/2/12	1002	Head reset **(POS 2)**
4/3/12	1036	2A EDG Inoperable (ESF testing)
4/4/12	0536	2A EDG Operable
4/4/12	1943	Unit 2 entered Mode 5
4/4/12	2003	Unit 1 reactor trip, loss of offsite power to both units. Loss of RHR and Spent Fuel Pool cooling due to loss of power. Both EDGs on both units automatically started and supplied the essential power busses.
4/4/12	2006	Started 2A RHR Pump to restore core cooling
4/4/12	2012	Unusual Event declared
4/4/12	2031	Started 2B Spent Fuel Pool Cooling Pump
4/4/12	approx 2045	Started raising Unit 2 Reactor Coolant System level. Level increased to approx 43%.
4/4/12	2122	TSC activated
4/4/12	2232	EOF activated
4/4/12	approx 2300	SSF D/G started
4/4/12	2306	SSF D/G declared inoperable due to operating at low voltage
4/5/12	0129	Offsite power restored to Unit 1 A-Train essential buss (1ETA)
4/5/12	0137	Offsite power restored to Unit 2 B-Train essential buss (2ETB)
4/5/12	0137	Unusual Event terminated
4/5/12	0138	1A EDG shutdown
4/5/12	0143	2B EDG shutdown
4/5/12	0236	Offsite power restored to Unit 2 A-Train essential buss (2ETA)
4/5/12	0245	2A EDG shutdown
4/5/12	0537	Offsite power restored to Unit 1 B-Train essential buss (1ETB)
4/5/12	0541	1B EDG shutdown
4/5/12	approx 0900	SSF D/G secured
4/5/12	approx 1200	Determined that LOOP was caused by a Zone G relay programming issue
4/5/12	1255	Started Unit 1 Condenser Circulating Water pump
4/5/12	approx 1400	Unit 2 outage schedule change to allow going to Loops Filled condition on Reactor Coolant System prior to performing B-Train ESF

		testing.
4/5/12	approx 1500	Restored cooling to Unit 1 Reactor Building and Reactor Coolant Pump motors
4/5/12	approx 1600	Started the "A" Auxiliary Electric Boiler
4/5/12	1648	Started the 1A Reactor Coolant Pump
4/5/12	approx 2100	Inspected 1T1B transformed and determined no damage
4/6/12	approx 0000	Entered Unit 1 containment and checked out 1D Reactor Coolant Pump Motor. Preliminary results indicate no damage to motor.
4/6/12	approx 0200	Determined problem with SSF D/G low voltage to be caused by power factor controller not being bypassed in isochronous mode.
4/6/12	approx 0800	Established Unit 1 condenser vacuum and restored condenser dump valves to service.
4/6/12	0802	SSF motor control center (1SLXG) powered from offsite power
4/6/12	approx 0900	Restored 1B Main Power
4/8/12	0258	Loops filled **(POS 1)**

Catawba Units 1 & 2 Power History



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Craver, Patti

From: Robles, Jesse
Sent: Monday, February 06, 2012 2:56 PM
Subject: New OpE COMM: Augmented Inspection - Wolf Creek Generating Station Loss of Offsite Power and Notification of Unusual Event

This email is being sent to notify recipients of a new posting on the @Operating Experience Community Forum.

Recipients are expected to review the posting for applicability to their areas of regulatory responsibility and consider appropriate actions. However, information contained in the posting is not tasking; therefore, no specific action or written response is required.

Summary

On January 13, 2012, Wolf Creek Generating Station experienced an automatic reactor trip and a loss of offsite power (LOOP). The site declared a Notification of Unusual Event (NOUE) (See EN 47590, PNO-IV-12-002, and PNO-IV-002A) as a result of the loss of offsite power. Several equipment issues were identified during the event, including ground alarms on an Emergency Diesel Generator (EDG), leaks on the Essential Service Water (ESW) system, an unexpected trip of the Turbine Driven Auxiliary Feedwater Pump (TDAFWP), and failure of a temporary diesel-driven fire pump (DFP). A Management Directive (MD) 8.3 evaluation was performed, and an Augmented Inspection Team (AIT) was sent to the site to gather additional information on the event.

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The posting may be reviewed at: **Augmented Inspection - Wolf Creek Generating Station Loss of Offsite Power and Notification of Unusual Event**

http://nrr10.nrc.gov/forum/forumtopic.cfm?selectedForum=03&forumId=AllComm&topicId=3720

This COMM is being posted to the following groups: ***All COMMS, Auxiliary Feedwater, Chemistry/Chemical Engineering, Containment (leakage, degradation, cooling system performance), ECCS, Electrical Power Systems, Emergency Diesel Generators, Emergency Preparedness, Fire Protection, Human Performance, HVAC, Inspection Programs, Instrumentation and Controls, Main Steam & Condensate/Feed Systems, New Reactors, Physical Security, Piping, Pump and Valve Performance, Safety Culture, SIT/AIT, Station Service Water Systems & Ultimate Heat Sink***

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C-3

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Craver, Patti

From: Haskell, Russell
Sent: Friday, February 10, 2012 12:27 PM
Cc: Lara, Julio; Brand, Javier; Powers, Dale; Taylor, Ryan
Subject: New OpE COMM: BYRON 2: NOUE/Special Inspection - DESIGN VULNERABILITY DISCOVERED IN ELECTRICAL DISTRIBUTION SYSTEM FOLLOWING REACTOR TRIP

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This email is being sent to notify recipients of a new OpE posting:

BYRON 2: NOUE/Special Inspection - DESIGN VULNERABILITY DISCOVERED IN ELECTRICAL DISTRIBUTION SYSTEM FOLLOWING REACTOR TRIP (click link)

> **Summary:** On January 30, 2012, unit 2 experienced a reactor trip from full power following an undervoltage condition on reactor coolant pump (RCP) electrical buses; tripping the RCPs. A walkdown of the switchyard identified a broken insulator stack connected between Station Auxiliary Transformer (SAT) a switchyard Revenue Meter. Two of four insulator sections were discovered on the ground in the switchyard. The broken insulator was originally connected to the 345 KV PHASE C line to the SATs, leading to the reactor trip condition.
>
> **MD 8.3** (revised) & Special Inspection **Charter**:
>
> *"Although for the actual event operators were successful in preventing a RCP seal induced LOCA from occurring, the risk is included in this revised analysis. The initial MD8.3 risk assessment determined the CCDP to be about 6.8E-6. The additional risk due to a small LOCA was calculated to be (CCDP) 3.06E-5 calculated as follows: 6.62E-3 * 0.21 * 0.022 = 3.06E-5. The additional risk due to a medium LOCA was inconsequential to the small LOCA risk, primarily due to the smaller probability of a medium LOCA resulting from RCP seal failure. Thus, the revised CCDP risk assessment is the sum of the above contributions or* ***3.7E-5****. This risk is in the overlap region between a Special Inspection and an Augmented Inspection Team (AIT)."..........read more at following link*

Distributed to the following OpE COMM groups: ***All Communications, ECCS, Electrical Power Systems, Emergency Diesel Generators, Emergency Preparedness, Human Performance, Inspection Programs, Instrumentation and Controls, New Reactors, Pump and Valve Performance, Quality Assurance and Vendor Issues, SIT/AIT, Station Service Water Systems & Ultimate Heat Sink***

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Craver, Patti

From: Haskell, Russell
Sent: Thursday, February 23, 2012 8:59 AM
Cc: Miller, Ilyne
Subject: New OpE COMM: PERRY 1: FINAL SIGNIFICANCE DETERMINATION OF WHITE FINDING ASSOCIATED WITH UNPLANNED DOSE RATES TO WORKERS

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This email is being sent to notify recipients of a new OpE posting:

PERRY 1: FINAL SIGNIFICANCE DETERMINATION OF WHITE FINDING (EA-11-148) ASSOCIATED WITH UNPLANNED DOSE RATES TO WORKERS (SPECIAL INSPECTION) (click link)

Summary:

This OpE COMM summarizes 2 similar events of licensee worker exposure to highly irradiated sources during maintenance activities on Nuclear Instrumentation. Both events led to reactive inspections. IOEB is issuing this summary of events as a reminder to staff of the hazards associated with similar activities. With the Spring 2012 Refueling Outage underway, the lessons learned included in this OpE COMM may be beneficial to some.

(April 2011) Perry 1: Final Significance Determination Of White Finding (EA-11-148) Associated With Unplanned Dose Rates To Workers MD 8.3 - Inspection Charter

(April 2011) Cooper: Unplanned Dose Rates To Workers During Extraction Of An Intermediate Range Monitor MD 8.3 - Inspection Charter

OpE COMM posting has been distributed to the following OpE groups: All Communications, Dose Assessment, Health Physics, Human Performance, Instrumentation and Controls, Safety Culture, Shutdown Risk, SIT/AIT, Spent Fuel Storage & Load Handling

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Thank you for reviewing and using Operating Experience.

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Craver, Patti

From: Robles, Jesse
Sent: Tuesday, April 10, 2012 3:22 PM
Subject: New OpE COMM: Augmented Inspection - San Onofre 3: Unit 3 Steam Generator Tubes Failed In-Situ Pressure Testing

This email is being sent to notify recipients of a new posting on the @Operating Experience Community Forum.

Recipients are expected to review the posting for applicability to their areas of regulatory responsibility and consider appropriate actions. However, information contained in the posting is not tasking; therefore, no specific action or written response is required.

Summary
On January 31, 2012, San Onofre Nuclear Generating Station (SONGS) Unit 3 experienced a steam generator tube leak that resulted in the unit being shutdown. During follow-up inspection of the Unit 3 steam generator tubes, the licensee discovered unexpected wear in both steam generators. These steam generators were manufactured by Mitsubishi Heavy Industries (MHI) and had been in service since the beginning of the operating cycle (approximately one year of power operation). The Unit 2 steam generators had been replaced during its previous refueling outage in 2009-2010.

This event and the subsequent failures of steam generator tubes during testing resulted in an Augmented Inspection Team (AIT) being sent to the site. A Confirmatory Action Letter (CAL) was issued to ensure that SONGS Unit 2 will not enter Mode 2 and SONGS Unit 3 will not enter Mode 4 until the cause of the abnormal wear is determined and the affected tubes are plugged. It also confirms that additional steam generator tube inspections will be performed during a mid-cycle outage.

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The posting may be reviewed at: **Augmented Inspection - San Onofre 3: Unit 3 Steam Generator Tubes Failed In-Situ Pressure Testing**

http://nrr10.nrc.gov/forum/forumtopic.cfm?selectedForum=03&forumId=AllComm&topicId=3798

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C6

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8

Craver, Patti

From: Giantelli, Joseph
Sent: Thursday, June 14, 2012 3:18 PM
To: Giantelli, Joseph
Subject: New OpE COMM Forum Posting: INDIAN POINT 3 - FAILURES OF MOTOR CUTOFF SWITCHES IN SAFETY RELATED CIRCUIT BREAKERS

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This e-mail is being sent to notify recipients of a new posting on the @Operating Experience Community Forum. Recipients are expected to review the posting for applicability to their areas of regulatory responsibility and consider appropriate actions. However, information contained in the posting is not tasking; therefore, no specific action or written response is required.

INDIAN POINT 3 - FAILURES OF MOTOR CUTOFF SWITCHES IN SAFETY RELATED CIRCUIT BREAKERS (Click this link to view the entire posting).

This OpE COMM provides a detailed description of the events associated with two reactor scram at River Bend Station that lead to the charter of an Augmented Inspection Team (AIT). This COMM will be updated as more information becomes available (i.e., AIT Report issued). Anyone with comments or questions regarding this COMM should contact: Steve Pannier (see contact information below).

This OpE COMM is being posted to the following groups and individuals: ***All Communications, Electrical Power Systems, Inspection Programs, Materials & Aging, New Reactors, Pump & Valve Performance and Quality Assurance & Vendor Issues***

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Thank You for reviewing and using Operating Experience.

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9

SERP Worksheet for SDP-Related Findings
Catawba Nuclear Station
Zone G Relay Modification - Unit 1 SDP

SERP Date: 07/18/2012 **EA No.:** 12-XXX

Licensee Name: Duke Energy Carolinas
Facility/Location: Catawba Nuclear Station
Docket No(s): 05000413
License No: NPF-35
Inspection Report No: 2012009
Date of Exit Meeting: June 18, 2012
Issue Sponsor: Region II
Deputy Director: Bill Jones **Division:** DRP
Branch Chief: Jonathan Bartley
Inspectors: Curt Rapp

Executive Summary:
Cornerstone Affected: □ IE ■ MS □ BI □ OR □ PR
Proposed Preliminary Results: □ White ■ Yellow □ Red □ Greater than Green
Summary of the Performance Deficiency: The licensee failed to follow the requirements of EDM 141, Procurement Specifications for Services, for providing appropriate design information to the vendor for programming the Unit 1 Zone G digital processors. Specifically, an "off-line" block for the generator underfrequency relay function was not programmed into the modification; therefore, any generator trip from high power would result in the opening of the Unit 1 switchyard breakers causing a loss of offsite power.

Summary of Significance Determination:
Provide a brief description of:
a. The Phase 1, Phase 2, and Phase 3 screening, logic process, and results
- Phase 1 - Finding represented a loss of system safety function (required Phase 2)
- Phase 2 - Finding screened as White under App. A, Table 3.7 - Loss of Offsite Power
- Phase 3 - ΔCDF = 4.7E-5, ΔLERF = 3.6E-6 (Yellow)

b. Influential Assumptions:
Window A (42 days): Both electrical buses ETA and ETB were aligned to Unit 2 supplies. If an event were to occur on Unit 1 during this time period, there would be minimal impact because both safety-related buses would maintain continuity of power. No quantification of the risk during this window was performed.
Window B (105 days): 1ETA re-aligned to a Unit 1 power supply. 1ETB aligned to Unit 2 supply. If an event were to occur on Unit 1 during this time period, the consequence of the performance deficiency would be a loss of ETA. All sequences evaluated. Recovery is not only possible, but highly likely due to the availability of the 'A' EDG and/or the cross-unit electrical feed.
Window C (40 days): Both safety-related busses being supplied from Unit 1. Any reactor trip from high power would cause the inadequate Zone G modification to divorce Unit 1 from the grid by opening up the switchyard feeder breakers. The following events/sequences were determined to cause a reactor trip and therefore needed to be evaluated: TRANS, LOACA, LOCHS, SGTR, LOMFW, LODCB, LOIA.
Window D (6 days): Both units were shutdown due to Technical Specification 3.0.3 issue with Control Room AC. No quantification of the risk during this window was performed.
Window E (105 days): Both safety-related busses being supplied from Unit 1. The risk analysis approach is identical to that of Window C.

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Recovery Actions: The analyst left all recovery actions at their nominal values assumed in the Catawba SPAR model. The only adjustment that was made was for recovery of the postulated failure of the ETA bus.

Standby Shutdown Facility (SSF): The SSF failed during this event but it is not being considered for the purposes of this analysis. This is because the analysis addresses only the risk of this performance deficiency, and any other PD or violation, if one is ultimately identified, will be treated separately and considered in isolation.

Ex-Core Sources: The analyst made no assessment of risk due to ex-core sources, e.g., fuel that might be damaged in the spent fuel pool due to the sustained loss of offsite power.

c. Dominant Cut-sets:

The dominant accident sequence for CDF is TRANS 21-18 and contributes 31% of the total internal events ΔCDF. The dominant accident sequence for LERF is SGTR 22-14 and contributes 69% of the total internal events ΔLERF.

d. Risk-insights:

This analysis was performed as a series of condition assessments. From the time that the performance deficiency was introduced to Unit 1 after the spring 2011 refueling outage until experienced a reactor trip and LOOP on April 4, 2012, various "risk windows" existed. The analyst identified each of these windows, determined how the performance deficiency would affect the plant, and then summed the risk for each of these windows.

e. Uncertainty and Sensitivity Studies:

Uncertainty

Upper bound for TRANS (ΔCDF) = 1.08E-4
Lower bound for TRANS (ΔCDF) = 6.4E-6
Upper bound for SGTR (ΔLERF) = 9.4E-6
Lower bound for SGTR (ΔLERF) = 6.7E-7

f. Contributions from External Events:

External Flooding - Would not cause an increase in the likelihood of a reactor trip without a LOOP. Therefore PD is present in both the base and non-conforming case.

Seismic - Same as above

Tornado - Same as above

Fire - The analyst performed a blended approach of qualitative and quantitative risk insights to demonstrate that it would not result in a change in color. Following the completion of this fire analysis, the licensee supplied risk information from their NFPA-805 transition efforts to support an estimate of the Fire Initiation Frequencies that cause a reactor trip. This number was estimated at 9E-2. Fire need not be considered any further for purposes of this analysis.

g. Potential Risk Contribution due to LERF:

NRC SPAR model for the Catawba plant did not have an ability to quantify LERF. Consequently, the analyst used the SDP Phase 2 notebooks to identify those core damage sequences that had LERF multipliers indicating that they could result in a large and prompt release to the public. Only SGTR and LOOP sequences had LERF multipliers. At the time of the completion of the analysis, the ΔLERF result was already greater-than 1E-6 (Yellow) based on the SGTR sequences and further effort was necessary to obtain the LOOP results. The analyst will continue to work to refine the estimate; also using the licensee's output from their CAFTA model to estimate LERF.

h. Total Estimated Change in Core Damage Frequency:

ΔCDF = 4.7E-5

i. Licensee's Risk Evaluation:

Comparison Between NRC and Licensee Results:

The analyst was not able to fully compare the licensee's results with the NRC results due to several factors:

- At the time of the completion of this analysis, the licensee had not yet finished their analysis of Window B and had not shared the results.
- For Windows C and E, it appeared that the TRANS scenarios that would become LOOP events

were not present in the dominant cutset results, which caused the analyst to question the completeness of the licensee's results.

- For Windows C and E, the licensee's initial results were dominated by accidents that the NRC's SPAR model was unable to quantify (e.g., LOCCW, external flooding rendering the SSF non-functional, and SORV).
- In addition, licensee concludes that in all cases, no core damage will occur with SSF available. This differs from some sequences in our model which do result in core damage. This discrepancy will need to be addressed.

j. Summary of Results and Impact:

The increase in core damage frequency (ΔCDF) for this event is 4.7×10-5 and the increase in large early release frequency (ΔLERF) for this event is 3.6×10-6, therefore, this condition should be treated as Yellow (i.e., ΔCDF greater than or equal to 10-5 and ΔLERF greater than or equal to 10-6).

Window	Start Date	End Date	Delta CDF	Delta LERF
A	11-Jun-11	23-Jul-11	0	0
B	23-Jul-11	5-Nov-11	3.08E-06	0
C	5-Nov-11	15-Dec-11	1.22E-05	1.01E-6
D	15-Dec-11	21-Dec-11	0	0
E	21-Dec-11	4-Apr-12	3.21E-05	2.65E-6
Totals =			**4.75E-05**	**3.67E-6**

Comparison Between Phase 2 and Phase 3 Results:

The SDP Phase 2 result was a White. However, there were various limitations in the ability of the Phase 2 sheets to represent the increase in risk. For example, the inspectors completed the LOOP worksheets but the PD would have resulted in a LOOP for every accident that caused a reactor trip. Also the inspectors used an IEL of zero, which was more appropriate for an event assessment as opposed to a condition assessment. And lastly, the licensee was given a recovery credit of one, which decreased the overall risk and was inappropriate especially for the short duration sequences where offsite power needed to be recovered in less than 2 hours. (Actual offsite power recovery was achieved during the event at ~ 5 hours after the LOOP.) When taken in total, these issues would explain the difference between the Phase 2 and Phase 3 results. (See Attachment 2 for details)

Summary of any Associated Apparent Violation:

Unit 1 TS 3.8.1 required in part that two qualified circuits between the offsite transmission network and the Onsite Essential Auxiliary Power System shall be operable when operating in MODES 1, 2, 3 or 4. The TS Action Statement for Condition C required that "with two offsite circuits inoperable, restore one offsite circuit to operable status within 24 hours."

Contrary to the above, from November 5, 2011, until April 4, 2012, while the unit was in MODE 1, two offsite circuits were inoperable due to the Zone G modification error and no action was taken to restore an offsite circuit to an operable status within 24 hours.

Details

A. Summary of Issue: (include a brief description of the root cause and licensee's corrective action(s), if available):

During the Unit 1 spring 2011 refueling outage, the licensee implemented a Zone G Relay Modification. The purpose of EC 89962, Zone G Relay Modification, was to replace electromechanical and static main generator relays with multifunction, microprocessor-based relays. These relays were designed to detect faults and other abnormal conditions and isolate any element of the power system that could jeopardize the continued operation or integrity of the remainder of the system. The original design used one train of protective relays mostly arranged in a two-out-of-two scheme for each protective relaying function. The new design provided two redundant trains of relays connected in a two-out-of-two scheme for each train. One of the functions was an underfrequency relay to protect the generator by opening the switchyard breakers connected to the generator. This function allowed for the unit to be isolated from the grid while the main generator continued to power station loads in cases where the grid experiences a significant disturbance such as a load rejection.

The design requirements included an "off-line" block for the generator underfrequency relay functions. The "off-line" condition was based on the position of generator breakers 1A and 1B. If these breakers were open, the unit was considered offline and the generator underfrequency function was blocked. The licensee determined that the "off-line" block was omitted by the relay vendor for the instantaneous underfrequency relay due to a programming error. The inspectors identified that the licensee missed multiple opportunities to discover the programming error during the testing phase of the modification. These opportunities were missed mainly because the licensee did not develop testing procedures from the original design specifications. Instead, the licensee used a calculation that was generated during the vendor's design portion of the modification as the basis for the testing procedures. Consequently, the programming error propagated through the rest of the implementation phase and was undetected either at the factory or during the post modification testing (PMT). Also, the relay replacement was considered a non-QA-1 modification. Therefore, much of the additional review and rigor in the licensee's design control process was not applicable to the modification.

The licensee used relays from two different vendors to avoid common cause failure issues. One of the relays also automatically blocked the underfrequency function based on relay input voltage. If generator output was below a specific relay input voltage, these relays would block the underfrequency function for that train. Any controlled shutdown would not result in a Unit 1 LOOP because the underfrequency function was blocked based on input voltage for that train preventing the two-out-of-two logic from opening the switchyard breakers. Only in the case of a turbine trip from high power would a LOOP result because the underfrequency setpoint was reached before the automatic block could occur.

This modification was also installed on Unit 2 during the spring 2012 refueling outage. The licensee used the same vendor to program the Unit 2 relays and the same PMT procedures used on Unit 1; therefore, the programming error also was undetected on Unit 2. If Unit 2 had been restarted and operated at power then a turbine trip would have resulted in a LOOP on Unit 2. However, the LOSP on Unit 1 allowed the licensee to identify and correct the programming error on Unit 2 prior to restart.

B. Statement of the Performance Deficiency:

The licensee failed to follow the requirements of EDM 141, Procurement Specifications for Services, for providing appropriate design information to the vendor for programming the Unit 1 Zone G digital processors. Specifically, an "off-line" block for the generator underfrequency relay function was not programmed into the modification; therefore, any generator trip from high power would result in the opening of the Unit 1 switchyard breakers causing a loss of offsite power. The PD was more than minor because it affected the availability and reliability of the Equipment Performance attribute and adversely affected the Mitigating Systems cornerstone objective in that an offsite power supply would not have been available to respond to expected operational transients.

C. Significance Determination Basis:

1. Reactor Inspection for IE, MS, BI cornerstones

a. Phase 1 screening logic:

Finding represented a loss of system safety function (requires Phase 2)

b. Phase 2 Risk Evaluation:

Finding screened as White under App. A, Table 3.7 - Loss of Offsite Power

(1) Select Phase 2 method used

□ SDP Interface (SAPHIRE Version 8) or

■ Phase 2 SDP Appendix used: __A__ (A through M)

(2) Preliminary Results: □ White ■ Yellow □ Red

(3) Provide the Phase 2 Evaluation (SDP Interface Report or SDP Appendix worksheet. (See Attached)

(4) If the preliminary risk significance determination based on Phase 2 SDP worksheet results is "Green" (1E-7) or higher significance, screen the risk contributions from external events (e.g., fire, seismic, and floods) that may add to the preliminary risk significance determination based on Phase 2 SDP worksheet results, using guidance in IMC 0609, Appendix A, Attachment 3. (See Phase 3)

c. Phase 3 Analysis:

Concisely address each of the analysis aspects that follow.

(1) The Phase 3 model revision and other PRA Tools used:

Model Used: Catawba SPAR Model Version 8.20 - Build #3 (INL model change to include offsite power recoveries for accident sequences other than LOOP)

Software Used: Saphire Version 8.0.7.17

(2) Influential Assumptions:

Window A (42 days): Both electrical buses ETA and ETB were aligned to Unit 2 supplies. If an event were to occur on Unit 1 during this time period, there would be minimal impact because both safety-related buses would maintain continuity of power. No quantification of the risk during this window was performed.

Window B (105 days): 1ETA re-aligned to a Unit 1 power supply. 1ETB aligned to Unit 2 supply. If an event were to occur on Unit 1 during this time period, the consequence of the performance deficiency would be a loss of ETA. All sequences evaluated. Recovery is not only possible, but highly likely due to the availability of the 'A' EDG and/or the cross-unit electrical feed.

Window C (40 days): Both safety-related busses being supplied from Unit 1. Any reactor trip from high power would cause the inadequate Zone G modification to divorce Unit 1 from the grid by opening up the switchyard feeder breakers. The following events/sequences were determined to cause a reactor trip and therefore needed to be evaluated: TRANS, LOACA, LOCHS, SGTR, LOMFW, LODCB, LOIA.

Window D (6 days): Both units were shutdown due to Technical Specification 3.0.3 issue with Control Room AC. No quantification of the risk during this window was performed.

Window E (105 days): Both safety-related busses being supplied from Unit 1. The risk analysis approach is identical to that of Window C.

Recovery Actions: The analyst left all recovery actions at their nominal values assumed in the Catawba SPAR model. The only adjustment that was made was for recovery of the postulated failure of the ETA bus.

Standby Shutdown Facility (SSF): The SSF failed during this event but it is not being considered for the purposes of this analysis. This is because the analysis addresses only the risk of this performance deficiency, and any other PD or violation, if one is ultimately identified, will be treated separately and considered in isolation.

Ex-Core Sources: The analyst made no assessment of risk due to ex-core sources, e.g., fuel that might be damaged in the spent fuel pool due to the sustained loss of offsite power.

(3) Calculation Discussion (SAPHIRE analysis results, SPAR-H evaluation):

The calculations performed by the analysis included the following:

- Application of exposure time for each time window
- Calculations of ΔCDF and ΔLERF for relevant time windows
- *SPAR-H calculation for EDG recovery for time window B*

(4) Analysis of Dominant Cut-sets / sequences:

The dominant accident sequence for CDF is TRANS 21-18 and contributes 31% of the total internal events ΔCDF. The dominant accident sequence for LERF is SGTR 22-14 and contributes 69% of the total internal events ΔLERF. The events and important component failures in TRAN Sequence 21-18 are:

- Reactor transient (TRANS) occurs,
- Offsite electrical power fails,
- *Emergency power system succeeds,*
- Auxiliary feedwater (AFW) fails,
- Primary feed and bleed fails, and
- Operators fail to recover offsite power within 2 hours.

(5) Sensitivity Analysis:

(a) Contributions of greatest uncertainty factors and impact on assumptions:

See discussion of comparison between NRC and licensee results.

There may be additional recovery credit. Specifically, EDG recovery actions, LOOP recovery actions, and cross-unit recovery actions are addressed in the model; however there may be additional recovery actions to available to restore power. Those have not yet been quantified at the completion of this Phase 3 analysis.

(b) The staff should bound the uncertainties, if possible, and through sensitivity analysis (quantitative and qualitative) state why they are conservative. Bounding an assumption between two reasoned limits and selecting an average value is acceptable. The SERP will judge whether the staff's arguments are reasonable and unbiased.

The analyst qualitatively evaluated the decrease in ΔCDF due to application of the site LOOP apportionment factor and the LOOP recovery HEP. This, if applied, may decrease risk to White at the lower bound.

With respect to the upper bound, the following events either could not be evaluated for the performance deficiency of concern, or were excluded from consideration by the risk analyst: SORV, LONSW, LOCCW and SLOCA. Those events that could not be evaluated may represent un-quantified risk that may add to the total result. The analyst does not believe the risk associated with these sequences would cause an increase in color to Red.

(6) Contributions from External Events:

External Flooding - Would not cause *an increase in the likelihood of a reactor trip* without a LOOP. Therefore PD is present in both the base and non-conforming case.

Seismic - Same as above

Tornado - Same as above

Fire - The analyst performed a blended approach of qualitative and quantitative risk insights to demonstrate that it would not result in a change in color. Following the completion of this fire analysis, the licensee supplied risk information from their NFPA-805 transition efforts to support an estimate of the Fire Initiation Frequencies that cause a reactor trip. This number was estimated at 9E-2. Fire need not be considered any further for purposes of this analysis.

(7) Potential Risk Contribution from LERF:
NRC SPAR model for the Catawba plant did not have an ability to quantify LERF. Consequently, the analyst used the SDP Phase 2 notebooks to identify those core damage sequences that had LERF multipliers indicating that they could result in a large and prompt release to the public. Only SGTR and LOOP sequences had LERF multipliers. At the time of the completion of the analysis, the ΔLERF result was already greater-than 1E-6 (Yellow) based on the SGTR sequences and further effort was necessary to obtain the LOOP results. The analyst will continue to work to refine the estimate; also using the licensee's output from their CAFTA model to estimate LERF.

(8) Total Estimated Change in Core Damage Frequency:
ΔCDF = 4.7E-5

(9) Licensee's Risk Evaluation:
Comparison Between NRC and Licensee Results:
The analyst was not able to fully compare the licensee's results with the NRC results due to several factors:

- At the time of the completion of this analysis, the licensee had not yet finished their analysis of Window B and had not shared the results.
- For Windows C and E, it appeared that the TRANS scenarios that would become LOOP events were not present in the dominant cutset results, which caused the analyst to question the completeness of the licensee's results.
- For Windows C and E, the licensee's initial results were dominated by accidents that the NRC's SPAR model was unable to quantify (e.g., LOCCW, external flooding rendering the SSF non-functional, and SORV).
- In addition, licensee concludes that in all cases, no core damage will occur with SSF available. This differs from some sequences in our model which do result in core damage. This discrepancy will need to be addressed.

(10) Summary of Results and Impact:
The increase in core damage frequency (ΔCDF) for this event is 4.7×10^{-5} and the increase in large early release frequency (ΔLERF) for this event is 3.6×10^{-6}, therefore, this condition should be treated as Yellow (i.e., ΔCDF greater than or equal to 10^{-5} and ΔLERF greater than or equal to 10^{-6}).

Window	Start Date	End Date	Delta CDF	Delta LERF
A	11-Jun-11	23-Jul-11	0	0
B	23-Jul-11	5-Nov-11	3.08E-06	0
C	5-Nov-11	15-Dec-11	1.22E-05	1.01E-6
D	15-Dec-11	21-Dec-11	0	0
E	21-Dec-11	4-Apr-12	3.21E-05	2.65E-6
Totals =			**4.75E-05**	**3.67E-6**

Comparison Between Phase 2 and Phase 3 Results:
The SDP Phase 2 result was a White. However, there were various limitations in the ability of the Phase 2 sheets to represent the increase in risk. For example, the inspectors completed the LOOP worksheets but the PD would have resulted in a LOOP for every accident that caused a reactor trip. Also the inspectors used an IEL of zero, which was more appropriate for an event assessment as opposed to a condition assessment. And lastly, the licensee was given a recovery credit of one, which decreased the overall risk and was inappropriate especially for the short duration sequences where offsite power needed to be recovered in less than 2 hours. (Actual offsite power recovery was achieved during the event at ~ 5 hours after the LOOP.) When taken in total, these issues would explain the difference between the Phase 2 and Phase 3 results. (See Attachment 2 for details)

d. Peer Review: George MacDonald

Summarize any unresolved issues identified by the reviewer.
N/A

e. References: (See Phase 3)

2. All Other Inspection Findings (not IE, MS, BI cornerstones)

Flowchart logic and full justification of assumptions used: N/A

Proposed preliminary or final color: N/A

D. Proposed Enforcement:

1. Regulatory requirement not met: TS 3.8.1

2. Proposed citation:
Unit 1 TS 3.8.1 required in part that two qualified circuits between the offsite transmission network and the Onsite Essential Auxiliary Power System shall be operable when operating in MODES 1, 2, 3 or 4. The TS Action Statement for Condition C required that "with two offsite circuits inoperable, restore one offsite circuit to operable status within 24 hours."

Contrary to the above, from November 5, 2011, until April 4, 2012, while the unit was in MODE 1, two offsite circuits were inoperable due to the Zone G modification error and no action was taken to restore an offsite circuit to an operable status within 24 hours.

E. Determination of Follow-up Review:

For White findings propose whether headquarters (NRR and/or OE) should review final determination letter before issuance. (For greater than White findings, review and concurrence by NRR and OE is required as discussed in Section 4b.)

Review and concurrence by NRR and OE

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SERP Worksheet for SDP-Related Findings
Catawba Nuclear Station
Zone G Relay Modification - Unit 2 SDP

SERP Date: 07/18/2012 **EA No.:** 12-XXX

Licensee Name: Duke Energy Carolinas
Facility/Location: Catawba Nuclear Station
Docket No(s): 05000414
License No: NPF-52
Inspection Report No: 2012009
Date of Exit Meeting: June 18, 2012
Issue Sponsor: Region II
Deputy Director: Bill Jones **Division:** DRP
Branch Chief: Jonathan Bartley
Inspectors: Curt Rapp

Executive Summary:
Cornerstone Affected: □ IE ■ MS □ BI □ OR □ PR
Proposed Preliminary Results: □ White □ Yellow □ Red ■ Greater than Green
Summary of the Performance Deficiency: The licensee failed to follow the requirements of EDM 141, Procurement Specifications for Services, for providing appropriate design information to the vendor for programming the Unit 1 Zone G digital processors. Specifically, an "off-line" block for the generator underfrequency relay function was not programmed into the modification; therefore, any generator trip from high power would result in the opening of the Unit 1 switchyard breakers causing a loss of offsite power. This would cause a loss of offsite power to Unit 2 because Unit 1 was aligned to provide offsite power to Unit 2.

Summary of Significance Determination:
Provide a brief description of:

a. The Phase 1, Phase 2, and Phase 3 screening, logic process, and results:
- Phase 1 - Finding represented a loss of system safety function (required Phase 2)
- Phase 2 -
  - Finding screened as Green under App. A, Table 3.7 - LOOP (at-power)
  - Finding screened as >Green under Appendix G, Worksheet 3 - PWR/LOOP (shutdown)
- Phase 3 - Finding resulted in ΔCDF of 4.11E-6, LERF uncertainty (Greater than Green)

b. Influential Assumptions:

Window A (At Power) (21 days): Only LOOP sequences were evaluated since any other event that originated in Unit 2 (e.g., LOMFW, TRANS) would progress normally.

Window B (POS1 TW-E) (64 hours): Both trains of RHR available for heat removal, RCS was filled and vented and the steam generators available, and 2B EDG was out-of-service for extensive maintenance. Plant-centered and/or switchyard-centered LOOP was credible. A weather-related or grid related LOOP was not considered

Window B (POS2 TW-E) (47 hours): Mode 6, reactor vessel (RV) level lowered below the flange for RV head removal, and both trains of RHR available. Plant-centered and/or switchyard-centered LOOP was credible. A weather-related or grid related LOOP was not considered

Window C (POS3) (18 days): Risk not evaluated because the potential loss of off-site or on-site emergency power was determined to be of low risk significance due to the high amount of inventory and/or recovery time available.

Window D (POS2 TW-L) (58 hours): Mode 6 and Mode 5, RV level lowered below the flange for RV head reinstallation, both trains of RHR available, and 2A EDG out-of-service

Window E (Actual Event): Plant conditions were identical to Window D, however the analyst

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evaluated the risk of the event by treating it as a precursor event.

c. Dominant Cut-sets:

The dominant accident sequence is a LOOP 19-02 during the At-Power window and contributes 92% of the total internal events ΔCDF.

d. Risk-insights:

The performance deficiency was identified as a result of the LOOP event on April 4, 2012. The condition potentially affected Unit 2 due to the crosstie of electrical power and therefore affected various plant modes and operational states. An approximation of the risk in the various shutdown conditions was obtained via the SDP Appendix G worksheets. Then the model was used for the analysis of the operating condition (Appendix A) to generate a ΔCDF and the results were summed. The event starts as a LOOP that affects both units with failure of both diesels and non-recovery of offsite power.

e. Uncertainty and Sensitivity Studies:

The analyst was unable to perform uncertainty calculations due to problems with the model.
Window A: As a sensitivity study, the analyst modified the offsite power recoveries at certain time periods (OEP-XHE-XL-NR02H, NR04H, and NR24H) due to their significance in the base case. This was an attempt to ensure that if somewhat more optimistic estimates were used for the offsite power recoveries, such as those in the McGuire SPAR model, that there would not be a significant change in the result. The non-conforming case in this scenario decreased from 7.43E-5 to 3.9E-5 and did not result in a color change.

f. Contributions from External Events:

External Flooding - Would not cause an increase in the likelihood of a reactor trip without a LOOP. Therefore PD is present in both the base and non-conforming case.
Seismic - Same as above
Tornado - Same as above
Fire - At the time of the completion of this analysis, the licensee had not supplied any risk information from their NFPA-805 transition efforts to support a detailed estimate of the Fire Initiation Frequencies that cause a reactor trip. However, the analyst performed a blended approach of qualitative and quantitative risk insights to demonstrate that it would not result in a change in color.

g. Potential Risk Contribution due to LERF:

NRC SPAR model did not have an ability to quantify LERF. Consequently, the analyst used the internal events CDF result and applied several "screenings" to determine those accident sequences that would contribute to LERF risk. The analyst filtered the SPAR model results for cutsets that involved failure to recover offsite power at the 1 hour and 2 hour timeframes, since these would constitute an event that rapidly evolved. Further screening was applied in order to consider only those sequences where both EDGs failed (i.e., SBO) and the TDAFW also failed (likely a high pressure sequence). The aforementioned sequences when quantified appeared to be greater than 1E-7 so were potentially White or higher on LERF.

h. Total Estimated Change in Core Damage Frequency:

4.11E-6

i. Licensee's Risk Evaluation:

Comparison Between NRC and Licensee Results:
At the time of completion of this analysis, the licensee had not yet provided their risk results. However, through discussions with Duke PRA personnel, the analyst determined that two of the significant differences between the NRC results and licensee results were:

- The apportionment factor used in the respective models, representing those LOOPs that are or become site-wide and affect both units. The NRC uses a value of 5.7E-1 whereas the licensee uses a value of 1.6E-1.
- The HEP associated with electrically cross-connecting the units is assumed to be 1E-1, whereas the licensee assumes 4.3E-2.

When taken together, if used, these factors would impact the NRC results and may result in a significant reduction in the risk estimate.

In addition, licensee concludes that in all cases, no core damage will occur with SSF available. This differs from some sequences in our model which do result in core damage. This discrepancy will need to be addressed.

j. Summary of Results and Impact:

Window	Individual Risk Results
A (At-Power)	3.8E-6
B (Shutdown – POS-1)	6.7E-9
C (Shutdown – POS-2)	1.43E-7
D (Shutdown – POS-3)	0
E (Shutdown – POS-2)	1.74E-7
F (Shutdown – Precursor Event)	1.4E-7
Total Result =	**4.11E-6**

Comparison Between Phase 2 and Phase 3 Results:

The Phase 2 results initially calculated by the inspectors for the At-Power window underestimated the risk because the LOOP frequency was not increased sufficiently and additional recovery credit was applied. The Phase 1 results for the Shutdown windows were initially much higher than the actual risk due to lack of refinement in the LOOP frequencies and the credit for EAC. No further reconciliation with the Phase 2 results is necessary.

Summary of any Associated Apparent Violation:

Unit 2 TS 3.8.1 required in part that two qualified circuits between the offsite transmission network and the Onsite Essential Auxiliary Power System shall be operable when operating in MODES 1, 2, 3 or 4. The TS Action Statement for Condition C required that "with two offsite circuits inoperable, restore one offsite circuit to operable status within 24 hours." Unit 2 TS 3.8.2 required in part that one qualified circuit between the offsite transmission network and the Onsite Essential Auxiliary Power distribution system shall be operable when operating in MODES 5, 6, and during movement of irradiated fuel assemblies. The TS Action Statement for Condition A required that "with one required offsite circuit inoperable, initiate action to restore required offsite power circuit to operable status immediately."

Contrary to the above, from February 18, 2012, until March 10, 2012, while the unit was operating in MODES 1-4, two offsite circuits were inoperable due to the Zone G modification error and no action was taken to restore an offsite circuit to an operable status within 24 hours. In addition, from March 10, 2012, until April 4, 2012, while the unit was operating in MODES 5-6, one offsite circuit was inoperable due to the Zone G modification error and no immediate action was taken to restore the circuit to an operable status.

Details

A. Summary of Issue: (include a brief description of the root cause and licensee's corrective action(s), if available):

During the Unit 1 spring 2011 refueling outage, the licensee implemented a Zone G Relay Modification. The purpose of EC 89962, Zone G Relay Modification, was to replace electromechanical and static main generator relays with multifunction, microprocessor-based relays. These relays were designed to detect faults and other abnormal conditions and isolate any element of the power system that could jeopardize the continued operation or integrity of the remainder of the system. The original design used one train of protective relays mostly arranged in a two-out-of-two scheme for each protective relaying function. The new design provided two redundant trains of relays connected in a two-out-of-two scheme for each train. One of the functions was an underfrequency relay to protect the generator by opening the switchyard breakers connected to the generator. This function allowed for the unit to be isolated from the grid while the main generator continued to power station loads in cases where the grid experiences a significant disturbance such as a load rejection.

The design requirements included an "off-line" block for the generator underfrequency relay functions. The "off-line" condition was based on the position of generator breakers 1A and 1B. If these breakers were open, the unit was considered offline and the generator underfrequency function was blocked. The licensee determined that the "off-line" block was omitted by the relay vendor for the instantaneous underfrequency relay due to a programming error. The inspectors identified that the licensee missed multiple opportunities to discover the programming error during the testing phase of the modification. These opportunities were missed mainly because the licensee did not develop testing procedures from the original design specifications. Instead, the licensee used a calculation that was generated during the vendor's design portion of the modification as the basis for the testing procedures. Consequently, the programming error propagated through the rest of the implementation phase and was undetected either at the factory or during the post modification testing (PMT). Also, the relay replacement was considered a non-QA-1 modification. Therefore, much of the additional review and rigor in the licensee's design control process was not applicable to the modification.

The licensee used relays from two different vendors to avoid common cause failure issues. One of the relays also automatically blocked the underfrequency function based on relay input voltage. If generator output was below a specific relay input voltage, these relays would block the underfrequency function for that train. Any controlled shutdown would not result in a Unit 1 LOOP because the underfrequency function was blocked based on input voltage for that train preventing the two-out-of-two logic from opening the switchyard breakers. Only in the case of a turbine trip from high power would a LOOP result because the underfrequency setpoint was reached before the automatic block could occur.

This modification was also installed on Unit 2 during the spring 2012 refueling outage. The licensee used the same vendor to program the Unit 2 relays and the same PMT procedures used on Unit 1; therefore, the programming error also was undetected on Unit 2. If Unit 2 had been restarted and operated at power then a turbine trip would have resulted in a LOOP on Unit 2. However, the LOSP on Unit 1 allowed the licensee to identify and correct the programming error on Unit 2 prior to restart.

B. Statement of the Performance Deficiency:

The licensee failed to follow the requirements of EDM 141, Procurement Specifications for Services, for providing appropriate design information to the vendor for programming the Unit 1 Zone G digital processors. Specifically, an "off-line" block for the generator underfrequency relay function was not programmed into the modification; therefore, any generator trip from high power would result in the opening of the Unit 1 switchyard breakers causing a loss of offsite power. This would cause a loss of offsite power to Unit 2 because Unit 1 was aligned to provide offsite power to Unit 2. The PD was more than minor because it affected the availability and reliability of the Equipment Performance attribute and adversely affected the Mitigating Systems cornerstone objective in that an offsite power supply would not have been available to respond to expected operational transients.

C. Significance Determination Basis:

1. Reactor Inspection for IE, MS, BI cornerstones

a. Phase 1 screening logic:

Finding represented a loss of system safety function (requires Phase 2)

b. Phase 2 Risk Evaluation:

Finding screened as Green under App. A, Table 3.7 - LOOP (at-power)
Finding screened as >Green under Appendix G, Worksheet 3 - PWR/LOOP (shutdown)

(1) Select Phase 2 method used

□ SDP Interface (SAPHIRE Version 8) or
■ Phase 2 SDP Appendix used: A and G (A through M)

(2) Preliminary Results: □ White □ Yellow □ Red ■ Greater than Green

(3) Provide the Phase 2 Evaluation (SDP Interface Report or SDP Appendix worksheet. (See Attached)

(4) If the preliminary risk significance determination based on Phase 2 SDP worksheet results is "Green" (1E-7) or higher significance, screen the risk contributions from external events (e.g., fire, seismic, and floods) that may add to the preliminary risk significance determination based on Phase 2 SDP worksheet results, using guidance in IMC 0609, Appendix A, Attachment 3. (See Phase 3)

c. Phase 3 Analysis:

Concisely address each of the analysis aspects that follow.

(1) The Phase 3 model revision and other PRA Tools used:

Model Used: Catawba SPAR Model Version 8.20
Software Used: Saphire Version 8.0.7.17

(2) Influential Assumptions:

The inadequately designed modification to the Unit 2 switchyard was in the process of being installed during the spring 2012 outage, when the LOOP event occurred. The increased risk to Unit 2 was due to Unit 1 modification to the Zone G protective relaying logic combined with the plant's procedure to cross-connect the safety-related electrical power during an outage. On February 4, 2012, the electrical bus 2ETA was aligned to Unit 1. On March 18, 2012, the electrical bus 2ETB was aligned to Unit 1 at that point thereby exposing Unit 2 to the increased risk because both trains were cross connected. On March 10, 2012, Unit 2 shutdown for a refueling outage, thus ending the at-power exposure period (19 days). On April 4, 2012, the LOOP occurred as a result of the Unit 1 reactor trip and turbine trip, which ended the shutdown exposure period (25 days total). The full exposure time (T) was used. Please refer to Attachment 9 for a simplified diagram of the exposure windows.

(3) Calculation Discussion (SAPHIRE analysis results, SPAR-H evaluation):

The calculations performed by the analysis included the following:

- Application of exposure time for each time window
- Modified shutdown Phase 2 calculations for the CCDP estimates for each window
- SPAR-H calculation for EDG recovery for time window F

(4) Analysis of Dominant Cut-sets / sequences:

The dominant accident sequence is a LOOP 19-02 during the At-Power window and contributes 92% of the total internal events ΔCDF. The events and important component failures in the at-power sequence are:
Loss of offsite power (LOOP) transient occurs due to the performance deficiency,

- Emergency AC Power fails from both EDGs
- Auxiliary Feedwater succeeds
- PORVs/SRVs remain closed during the event
- Standby Shutdown Facility cooling to the RCP seals succeeds
- Failure to recover offsite power in 2 hours, leading to core damage

(5) Sensitivity Analysis:

(a) Contributions of greatest uncertainty factors and impact on assumptions:
See discussion of comparison between NRC and licensee results.
There may be additional recovery credit. Specifically, EDG recovery actions, LOOP recovery actions, and cross-unit recovery actions are addressed in the model; however there may be additional recovery actions to available to restore power. Those have not yet been quantified at the completion of this Phase 3 analysis.

(b) The staff should bound the uncertainties, if possible, and through sensitivity analysis (quantitative and qualitative) state why they are conservative. Bounding an assumption between two reasoned limits and selecting an average value is acceptable. The SERP will judge whether the staff's arguments are reasonable and unbiased.
The analyst qualitatively evaluated the decrease in ΔCDF due to application of the site LOOP apportionment factor and the LOOP recovery HEP. If applied, the risk may decrease to a high E-7 result.
Analyst attempted to determine the upper bound for this analysis by taking all LOOP sequences in the results and applying a 1.0 LERF multiplier with a result of ΔLERF 2.8E-6 (Yellow for LERF). This would be overly conservative since only some LOOP sequences are high pressure and LERF multipliers may be less than 1.0.

(6) Contributions from External Events:
External Flooding - Would not cause an increase in the likelihood of a reactor trip without a LOOP. Therefore PD is present in both the base and non-conforming case.
Seismic - Same as above
Tornado - Same as above
Fire - At the time of the completion of this analysis, the licensee had not supplied any risk information from their NFPA-805 transition efforts to support a detailed estimate of the Fire Initiation Frequencies that cause a reactor trip. However, the analyst performed a blended approach of qualitative and quantitative risk insights to demonstrate that it would not result in a change in color.

(7) Potential Risk Contribution from LERF:
NRC SPAR model did not have an ability to quantify LERF. Consequently, the analyst used the internal events CDF result and applied several "screenings" to determine those accident sequences that would contribute to LERF risk. The analyst filtered the SPAR model results for cutsets that involved failure to recover offsite power at the 1 hour and 2 hour timeframes, since these would constitute an event that rapidly evolved. Further screening was applied in order to consider only those sequences where both EDGs failed (i.e., SBO) and the TDAFW also failed (likely a high pressure sequence). The aforementioned sequences when quantified appeared to be greater than 1E-7 so were potentially White or higher on LERF. Further work is needed to refine this number.

(8) Total Estimated Change in Core Damage Frequency:
4.11E-6

(9) Licensee's Risk Evaluation:
Comparison Between NRC and Licensee Results:

At the time of completion of this analysis, the licensee had not yet provided their risk results. However, through discussions with Duke PRA personnel, the analyst determined that two of the significant differences between the NRC results and licensee results were:

- The apportionment factor used in the respective models, representing those LOOPs that are or become site-wide and affect both units. The NRC uses a value of 5.7E-1 whereas the licensee uses a value of 1.6E-1.
- The HEP associated with electrically cross connecting the units is assumed to be 1E-1, whereas the licensee assumes 4.3E-2.

When taken together, if used, these factors would impact the NRC results and may result in a significant reduction in the risk estimate.

In addition, licensee concludes that in all cases, no core damage will occur with SSF available. This differs from some sequences in our model which do result in core damage. This discrepancy will need to be addressed.

(10) Summary of Results and Impact:

Summation of Results:

Window	Individual Risk Results
A (At-Power)	3.8E-6
B (Shutdown – POS-1)	6.7E-9
C (Shutdown – POS-2)	1.43E-7
D (Shutdown – POS-3)	0
E (Shutdown – POS-2)	1.74E-7
F (Shutdown – Precursor Event)	1.4E-7
Total Result =	**4.11E-6**

Comparison Between Phase 2 and Phase 3 Results:

The Phase 2 results initially calculated by the inspectors for the At-Power window underestimated the risk because the LOOP frequency was not increased sufficiently and additional recovery credit was applied. The Phase 1 results for the Shutdown windows were initially much higher than the actual risk due to lack of refinement in the LOOP frequencies and the credit for EAC. No further reconciliation with the Phase 2 results is necessary.

d. Peer Review: Rudy Bernhard

Summarize any unresolved issues identified by the reviewer.

N/A

e. References: (See Phase 3)

2. All Other Inspection Findings (not IE, MS, BI cornerstones)

Flowchart logic and full justification of assumptions used: N/A

Proposed preliminary or final color: N/A

D. Proposed Enforcement:

1. Regulatory requirement not met: TS 3.8.1 and TS 3.8.2

2. Proposed citation:

Unit 2 TS 3.8.1 required in part that two qualified circuits between the offsite transmission network and the Onsite Essential Auxiliary Power System shall be operable when operating in MODES 1, 2, 3 or 4. The TS Action Statement for Condition C required that "with two offsite circuits inoperable, restore one offsite circuit to operable status within 24 hours." Unit 2 TS 3.8.2 required in part that one qualified circuit between the offsite transmission network and the Onsite Essential Auxiliary Power distribution system shall be operable when operating in MODES 5, 6, and during movement of irradiated fuel assemblies. The TS Action Statement for Condition A required that "with one required offsite circuit inoperable, initiate action to restore required offsite power circuit to operable status immediately."

Contrary to the above, from February 18, 2012, until March 10, 2012, while the unit was operating in MODES 1-4, two offsite circuits were inoperable due to the Zone G modification error and no action was taken to restore an offsite circuit to an operable status within 24 hours. In addition, from March 10, 2012, until April 4, 2012, while the unit was operating in MODES 5-6, one offsite circuit was inoperable due to the Zone G modification error and no immediate action was taken to restore the circuit to an operable status.

E. Determination of Follow-up Review (as needed)

For White findings propose whether headquarters (NRR and/or OE) should review final determination letter before issuance. (For greater than White findings, review and concurrence by NRR and OE is required as discussed in Section 4b.)

Review and concurrence by NRR and OE if final determination is greater than White

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Gibson, Lauren

From: Broaddus, Doug
Sent: Wednesday, July 18, 2012 10:51 AM
To: Hall, Randy; Paige, Jason; Gibson, Lauren
Subject: FW: SONGS AIT Report

FYI

From: Lund, Louise
Sent: Wednesday, July 18, 2012 10:26 AM
To: Boger, Bruce; Leeds, Eric; Dorman, Dan; Evans, Michele; Markley, Michael; Broaddus, Doug; Nieh, Ho; Hiland, Patrick
Subject: RE: SONGS AIT Report

FYI - Here's the list from the AIT report -

Ten items requiring additional followup are documented as unresolved items:

URI 1	Adequacy of the Trip/Transient and Event Review Procedure
URI 2	Evaluation of Unit 3 Vibration and Loose Parts Monitoring System Alarms
URI 3	Evaluation of Retainer Bars Vibration during the Original Design of the Replacement Steam Generators
URI 4	Evaluation of Changes in Dimensional Controls during the Fabrication of Unit 2 and Unit 3 Replacement Steam Generators
URI 5	Shipping Requirements not in Accordance with Industry Standards
URI 6	Shipping Requirements not in Accordance with Design and Fabrication Specifications
URI 7	Unit 3 Steam Generator 3E0-88 Stresses Related to Handling
URI 8	Non-Conservative Thermal-Hydraulic Model Results
URI 9	Evaluation of the Effects of Divider Plate Weld Repairs in Unit 3 Replacement Steam Generators
URI 10	Evaluation of Departure of Method of Evaluation for 10 CFR 50.59 Processes

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From: Boger, Bruce
Sent: Wednesday, July 18, 2012 10:11 AM
To: Leeds, Eric; Dorman, Dan; Evans, Michele; Lund, Louise; Markley, Michael; Broaddus, Doug; Nieh, Ho; Hiland, Patrick
Subject: SONGS AIT Report

Elmo called me to let us know he plans to issue the SONGS AIT Report today. It has 10 URIs, one of which relates to 50.59 and the licensee's use of several computer codes (is this a change in methodology?). The report also highlights the unprecedented steam generator tube vibration.

C 3

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Gibson, Lauren

From: Hall, Randy
Sent: Thursday, July 19, 2012 3:05 PM
To: Evans, Michele; Lund, Louise; Broaddus, Doug
Cc: Sebrosky, Joseph; Markley, Michael; Miranda, Samuel; Kulesa, Gloria; Hiland, Patrick; Cheok, Michael; Lubinski, John; Ruland, William; Davis, Jack; Karwoski, Kenneth; Murphy, Emmett; Johnson, Andrew; Parks, Benjamin; Beaulieu, David; Cartwright, William; Thurston, Carl; Paige, Jason; Gibson, Lauren
Subject: SONGS AIT Report

Here is a link to the AIT inspection report for the SONGS SG tube issues, dated July 18, 2012.

http://www.nrc.gov/info-finder/reactor/songs/ML12188A748.pdf

Randy Hall, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation
USNRC
(301) 415-4032
Randy.Hall@nrc.gov

C-11

13

Craver, Patti

From: Pannier, Stephen
Sent: Thursday, August 16, 2012 10:17 AM
Subject: Updated OpE COMM: AUGMENTED INSPECTION - SAN ONOFRE 3: UNIT 3 STEAM GENERATOR TUBES FAILED IN-SITU PRESSURE TESTING

Information Security Reminder: OpE COMMs contain preliminary information in the interest of timely internal communication of operating experience. OpE COMMs may be pre-decisional and may contain sensitive/proprietary information. They are not intended for distribution outside the agency.

This e-mail is being sent to notify recipients of an **update** to a posting on the @Operating Experience Community Forum. Recipients are expected to review the posting for applicability to their areas of regulatory responsibility and consider appropriate actions. However, information contained in the posting is not tasking; therefore, no specific action or written response is required.

AUGMENTED INSPECTION - SAN ONOFRE 3: UNIT 3 STEAM GENERATOR TUBES FAILED IN-SITU PRESSURE TESTING (Click this link to view the entire posting).

This OpE COMM update adds hyperlinks to the San Onofre Nuclear Generating Station (SONGS) Confirmatory Action Letter (CAL) and to the SONGS - NRC Augmented Inspection Team Report. The COMM update also provides Construction Operating Experience highlighting underlying issues associated with the SONGS event which are applicable to new reactors design. This COMM will continue to be updated as more information becomes available (i.e., resolution of applicable URI's). Anyone with comments or questions regarding this COMM should contact: Steve Pannier (see contact information below).

This OpE COMM update notice is being sent to the following groups and individuals: ***All Communications, Chemistry/Chemical Engineering, Dose Assessment, Emergency Preparedness, Health Physics, Human Performance, Inspection Programs, Main Steam & Condensate/Feed Systems, Materials/Aging, New Reactors, Piping, Power Uprate, Quality Assurance and Vendor Issues, RCPB Leakage, Safety Culture, SIT/AIT, Steam Generators, Welding/Non-Destructive Examination***

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For more information on the Reactor OpE Program, please visit our OpE Gateway at: http://nrr10.nrc.gov/ope-info-gateway/index.html

Thank You for reviewing and using Operating Experience.

Steve Pannier
Reactor Systems Engineer
US Nuclear Regulatory Commission
NRR/DIRS/IOEB
O-7E04; MS O-7C02A
301-415-4083
Stephen.Pannier@nrc.gov

C12

19

Craver, Patti

From: Pannier, Stephen
Sent: Thursday, August 16, 2012 10:38 AM
Subject: Updated OpE COMM: River Bend Station - Reactor Scram with Loss of Normal Service Water leads to an Augmented Inspection (AIT)

Information Security Reminder: OpE COMMs contain preliminary information in the interest of timely internal communication of operating experience. OpE COMMs may be pre-decisional and may contain sensitive/proprietary information. They are not intended for distribution outside the agency.

This e-mail is being sent to notify recipients of an **update** to a posting on the @Operating Experience Community Forum. Recipients are expected to review the posting for applicability to their areas of regulatory responsibility and consider appropriate actions. However, information contained in the posting is not tasking; therefore, no specific action or written response is required.

River Bend Station - Reactor Scram with Loss of Normal Service Water leads to an Augmented Inspection (AIT) (Click this link to view the entire posting).

This OpE COMM update adds a hyperlink to the River Bend Station - NRC Augmented Inspection Team Report. The COMM update also provides a summary of the unresolved items (URI's) requiring follow-up inspection to determine the existence and significance of any associated performance deficiencies. This COMM will continue to be updated as more information becomes available (i.e., resolution of applicable URI's). Anyone with comments or questions regarding this COMM should contact: Steve Pannier (see contact information below).

This OpE COMM update notice is being sent to the following groups and individuals: ***All Communications, Auxiliary Feedwater, ECCS, Electrical Power Systems, Emergency Preparedness, Fire Protection, Flood Protection & Missiles, Human Performance, Inspection Programs, Instrumentation and Controls, Main Steam & Condensate/Feed Systems, Materials/Aging, New Reactors, Quality Assurance and Vendor Issues, Safety Culture, SIT/AIT, Station Service Water Systems & Ultimate Heat Sink***

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For more information on the Reactor OpE Program, please visit our OpE Gateway at: http://nrr10.nrc.gov/ope-info-gateway/index.html

Thank You for reviewing and using Operating Experience.

Steve Pannier
Reactor Systems Engineer
US Nuclear Regulatory Commission
NRR/DIRS/IOEB
O-7E04; MS O-7C02A
301-415-4083
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Licensee's point:

- The Fort Calhoun incident was specific to their "spare" housings which were in a high oxygenated environment
- After years of testing Fort Calhoun did not identify this conditions in any other of their weld #5s
- The Palisades weld #3 issue involved very high stresses due to the environment it was exposed to (close to a crud trap, heavily grinded)
- The AREVA report did not provide additional guidance related to the susceptibility of weld overlays that would lead one to conclude they should be looked at.
- All industry experience pointed to the necessity of a heavy oxygenated environment or heavy stresses which weld #5 did not posses.
- The understanding at the time was that high levels of oxygen provide the primary catalyst for TGSCC in nuclear plant primary coolant system, and they concluded this condition did not exist in weld #5
- Weld residual stresses alone if of sufficient magnitude, can drive a crack through-wall in a TGSCC environment (the licensee considered Weld #5 to have very low residual stresses due to it being an overlay and the location provided an easier environment for welding and the weld #5 had a smooth finish)
- The 2001 root cause states that in comparison to the weld #3 cracking at Palisades, Fort Calhoun had essentially the same stress, very similar material, probably very similar chloride concentration, a lower temperature and a significantly higher dissolved oxygen concentration. This supports the licensee's position that oxygen played a heavy role in the cracking of weld #5 at Fort Calhoun a condition that was not believed to exist at Palisades.
- Based on CRDM-21 taking 29 years to fail, the licensee concluded the current housings should last through the end of life of the plant, especially with the design changes made which would theoretically reduce the susceptibility of TGSCC for Weld #3 . Also at the point these decisions were being, made the licensee was intending to replace the Head as well as all the housings in 2006, an activity which was not performed due to the economic downturn.
- There was very little OE and industry guidance at this point in time and the information available discounted Weld #5 in a vented CRDM as being an area for concern.
- Weld #5 is not covered by the ASME code.

NRC's point

- The root cause points out that weld #5 is exposed to the same environment as weld #3
- The Fort Calhoun issue proved that the weld overlay could obtain a flaw that would propagate through wall.
- After changing the design they did not perform a susceptibility analysis based on the resulting stresses on all the welds as they had done prior to deciding to replace all the housings (still confirming this, requested documentation if it exists)
- The root cause points out that manufacturing played a heavy role in this issue yet in 2001 fabrication restrictions were not applied to weld #5 as they were to weld #4 and #3.
- In 2001 the root cause was determined to be TGSCC that occurred as a result of a susceptible material existing in an enabling environment under adverse stress conditions, and by not analyzing the stress conditions of Weld #5 in the post 2001 design the licensee could not make the determination that these characteristics did not apply to weld #5 (still confirming, requested supporting documentation to refute)
- Machining was categorized as an issue during the 2001 root cause and it was confirmed during the 2012 testing that machining was performed on CRDM 24 weld #5 that wasn't taken into consideration as far as what the residual stresses in the weld could be.
- By identifying weld #5 as less susceptible does not eliminative the susceptibility all together which means it should have been considered when developing an inspection plan.
- The licensee did not consider the effects of cold working on the stresses in weld #5.
- The 2001 root cause states that in comparison to the weld #3 cracking at Palisades, Fort Calhoun had essentially the same stress, very similar material, probably very similar chloride concentration, a lower temperature and a significantly higher dissolved oxygen concentration. This supports our point because it is mentioned on numeral occasions that the factors necessary for TGSCC propagation are environment, susceptible material and stress, which in accordance with this statement are very similar between Weld #3 and Weld #5.
- The licensee identified using Alloy 600 would provide resistance to IGSCC and TGSCC yet did not go with this option due to cost.
- In the organizational/programmatic weakness section of the 2012 root cause the licensee states (and tags as associated with RC1 and CC1) The 1991 Fort Calhoun OE was not adequately utilized to include inspections of the housing ID weld buildups.

My current position:

Based on my review of the 2001 and 2012 root causes (pending additional information requested) my position is that the licensee failed to perform evaluations and analyses of the stresses specifically for the weld overlay for the CRDM design installed in 2001 resulting in a failure to take corrective actions necessary to prevent recurrence.

1. Unresolved Item Crack growth rate
    a. The NRC notes that the conclusions section of the root cause report includes a conclusion from a B&W which indicates that the fracture surface contains beach marks but that it cannot be determined whether those beach marks relate to refueling outages or more frequent events, e.g., occurring over 24 months.
    b. Based on item 1 the NRC notes that the Root Cause appropriately addresses several potential corrosion rates.
    c. The NRC finds this your approach to addressing the uncertainty in corrosion rate acceptable because the inspection interval contained in your inspection plan bounds all the corrosion rates discussed, i.e., the current inspection interval is one refueling outage and all crack growth rates proposed require longer than one outage for a crack to grow from non detectible to through wall
    d. The NRC does note some weaknesses in your root cause / inspection plan related to crack growth rate These are:
        i. Crack growth rate discussion bases one crack growth rate on 6 outages occurring in 11 months. Given that not all of these outages resulted in pressure or heat up cool down cycles, a more appropriate time interval would be 24 months
        ii. Given the above cited error in the time interval for one of the crack growth rate calculations, the value cited in the root cause report is over estimated. When the correct time period is used, the calculated values is consistent with crack growth rates from other events. This crack growth rate should not be characterized as "overly conservative" as is currently the case
        iii. The crack growth rate based on refueling cycles appears to be under estimated. NRC inquires into the operation of the plant revealed that the CRDM housings were vented, allowing oxygen to enter, at least one time in addition to refueling outages. This was for an outage to replace seals on a reactor coolant pump. Additional opportunities for oxygen ingress may occur each time a seal housing is replaced. Based on these observations, the crack growth rate identified is understated by at least one refueling outage.
        iv. Although the inspection plan is designed to be bounding to the most rapid crack growth rate considered, the only mention of crack growth rate in the inspection plan is that a through wall crack requires 4 – 5 cycles to grow. This statement is inconsistent with the root cause evaluation and is considered non conservative. This statement could inadvertently result in a revision of the inspection interval to a non conservative value.

    e. Another potential weakness I would like to point out from your root cause report is the inconsistencies from one section to another. You point out that you haven't been able to pinpoint the exact cause yet you make the statement that CRD-24 is unique. This may be true but without verification, making that declarative statement isn't necessarily accurate and it has the potential to limit what you're looking at and for. I understand you need to draw the line somewhere but I would suggest you keep in mind that you don't have a smoking gun when contemplating what options you have going forward.

2. Unresolved Item Failure to prevent recurrence and technical specifications violation for operating with pressure boundary leakage.
    a. Based on our review of your 2001 root cause report, 2012 root cause report, various vendor documents and interviews with your staff we identified a performance deficiency
    b. PD Failure to recognize the susceptibility of weld # 5 to TGSCC and therefore not apply the level of scrutiny and corrective actions to this weld resulting in a failure to prevent recurrence of leakage in the CRDM housing due to TGSCC.
    c. More than minor because it adversely affects the initiating events cornerstone objective for not limit the likelihood of events that upset plant stability, specifically the *cornerstone attribute of equipment performance*.
    d. This is a violation of 10 CFR appendix B Criterion XVI Corrective Actions for failure to prevent recurrence of a significant condition adverse to quality. I can't call this a NCV as of yet, until it has been entered into your corrective action program.
    e. Because we concluded there was a failure to prevent recurrence we will not be recommending discretion be granted for the technical specification violation for operating with pressure boundary leakage for greater than the LCO specified time, but rather than issuing to separate violations, we would combine the two into one violation with two examples.
    f. The performance deficiency screened as green after screening under the initiating events cornerstone because we answered no to the question if after reasonable assessment of degradation, could the finding result in exceeding the RCS leak rate for a small LOCA and could the finding have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. Basically because of the nature of the cracking and your site procedures we believe it would be detected and corrected prior to reaching the small break LOCA limits.
    g. We are not recommending a cross cutting aspect be applied to this performance deficiency because *it occurred more than three years ago* (11 years ago).

3. Violation Failure to follow the root cause procedure
    a. Through our review we raised concerns on the exclusion of welds 3&4 from your generic implications section in the root cause report and inspection plan. And

though we understand you are compiling additional information to provide to us the position I am leaving with is as follows.

b. PD Failure to adequately consider welds 3 and 4 in the generic implications portion of the root cause report and therefore provide justification for why no additional corrective actions associated with these welds are needed.
c. What our concern is with this issue is that you may again be applying a narrow focus for what the potential of this cracking is. By essentially looking for like for like scenarios rather than considering what other portions of the components could be subject to the same or similar factors and whether the factors that discount them are valid based on research, analysis, review and or inspection.
d. We consider this a violation of 10 CFR Appendix B Criterion V, for failure to follow procedures. Specifically the root cause procedure which requires you establish corrective actions for valid generic implications and if no corrective actions are proposed THEN document the rational.
e. We categorized this issue as more than minor because if left uncorrected it has the potential to lead to a more significant safety concern Specifically if you don't have adequate justification for not including welds 3 and 4 in your generic implications section and have adequate justification for not taking corrective actions to address the potential generic concern it may result in another through wall leak.
f. The performance deficiency screened as green after screening under the initiating events cornerstone because we answered no to the question if after reasonable assessment of degradation, could the finding result in exceeding the RCS leak rate for a small LOCA and could the finding have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. Basically because of the nature of the cracking and your site procedures we believe it would be detected and corrected prior to reaching the small break LOCA limits.
g. We are proposing a cross cutting aspect in the area of Human Performance, decision making for non-conservative assumptions. Specifically we believe you did not use non conservative assumptions when electing not to include welds 3 and 4 as a potential generic implication. But going forward we would be open to discussions on what you think the potential cross cutting aspect associated with this finding is.
h. We understand what your position is on this issue and as mentioned before we will be reviewing the additional documentation you are providing us and would change the characterization of this issue as necessary depending on the results of our review. If the violation doesn't change and after the report is issued you want to contest this issue it will be delineated in the inspection report the process to use to do that.
i. The formal exit of this issue will be during the resident's quarterly exit and we will be in communication with you to inform you of any changes regarding this issue.

1. URI Crack growth rate:

Dave Alley and I performed a follow-up inspection to determine if the assumptions you made were conservative and the planned actions bounded those conservative assumptions. We reviewed a variety of documents associated with crack growth and inspection intervals and noted the following statements included in the root cause report and vendor documents related to the determination of the appropriate crack growth rate.

1. The lab conducting the failure analysis concluded, it could not be conclusively determined if the beach marks corresponded to refueling outages, (i.e., 18 month cycle) or shorter periods as occurred during outages over the past 24 months
2. Palisades CRDM housing 21 leaked at weld 3 in 2001. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 failure. In calculating the crack growth rate of this crack, one contractor utilized an interval between beach marks which is much shorter than refueling outages. The intervals used are consistent with plant thermal cycles in which oxygen may or may not have been admitted into the CRDMs.
3. A spare CRDM housing at Ft Calhoun leaked at weld 5 in 1990. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 failure. In calculating the crack growth rate of this crack, Ft Calhoun stated that the beach marks were related to refueling cycles. Ft Calhoun also performed calculations indicating that the oxygen level at the location of the flaw did not change with time (including in response to refueling outages) because the spare CRDM housing was not vented. Ft Calhoun's evaluation indicated that oxygen levels at the vicinity of the crack would have begun to decline through diffusion and convection had the intervals between outages been much longer than 18 months. This is interpreted to mean that the beach marks at Ft Calhoun are in response to pressure/thermal cycles.
4. In at least one instance Palisades needed to repair the seals on a reactor coolant pump at a time other than an outage. This necessitates draining some of the water from the reactor coolant system and venting (admitting oxygen into) the CRDM housing. This represents an additional oxygen ingress event not included when determination of time to cracking is based on refueling outages.
5. In its inspection plan Palisades states that it will inspect all CRDM housings over the next 4 refueling outages, i.e., the interval between inspections is 1 refueling outage

Based on the above review, we noted that there are certain non conservative statements contained in the Root Cause Report and the inspection plan. These include:

1. The crack growth rate based on refueling outages is understated. If oxygen ingress is related to beach marks, given the oxygen ingress event which occurred to repair reactor coolant pump seals, 6 beach marks would occur in a maximum of 5 refueling intervals rather than the 6 refueling intervals that were used to calculate the crack growth rate in the root cause report.
2. The crack growth rate based on heat up and cool down cycles is overstated. The value in the root cause is based on 11 months. While 6 shutdowns did occur at the plant in 11 months several of these events did not result in pressure/temperature changes of the reactor coolant system. The appropriate time frame is 24 months rather than 11.

3. The inspection plan contains a non conservative statement: "However, once the crack has been initiated it propagates over 4 to 5 operating cycles prior to going through wall." While this statement does reflect one of the proposed theories for crack growth, sufficient evidence to demonstrate reasonable assurance that this theory is correct, and thereby overcome the non-conservatism of this statement, does not exist.

Despite the existence of the non conservatisms stated above, we concluded:

1. Sufficient evidence to conclusively determine the rate of crack growth does, and will not exist.
2. Crack growth based on pressure/temperature cycles is the most conservative of the potential crack growth mechanisms. In the absence of reasonable assurance of the correctness of less conservative mechanisms, through wall crack growth in 2 years must be utilized.
3. The licensee has not formally committed to any of the crack growth mechanisms discussed.
4. Your inspection program includes inspections in each of the next 4 outages. This inspection interval, once per outage, bounds all the crack growth mechanisms considered.

We find this approach to inspection to be both acceptable and sufficient justification to close this URI.

2. Unresolved Item Failure to prevent recurrence and technical specifications violation for operating with pressure boundary leakage.
    a. When the leak was identified in 2001 various corrective actions were applied to prevent recurrence. These corrective actions were limited to pressure boundary welds and the need for corrective actions related to weld 5 was not considered.
    b. Based on our review of your 2001 root cause report, 2012 root cause report, various vendor documents and interviews with your staff we identified you failed to eliminate one or more of the necessary factors at the weld build-up area to preclude TGSCC in the replacement housings. Specifically
        i. The 2001 root cause report documented weld 5 is exposed essentially to the same environment as weld 3.
        ii. Fabrication restrictions to prohibit grinding were not applied to the weld build-up region which promoted residual tensile stresses on the ID of the CRDM surface.
        iii. Material was changed from type 347 to type 316 stainless steel which are essentially equal as far as resistance to TGSCC
    c. Based on the recurrence of through wall leakage in the CRDM housings that occurred at the weld build-up region of the CRDM housings by TGSCC we concluded that the actions taken in 2001 were not adequate because the appropriate actions to preclude recurrence were within your ability to foresee and implement.
    d. We identified a performance deficiency for failure to prevent recurrence of a significant condition adverse to quality resulting in a non-compliance with the TS.

e. More than minor because it adversely affects the initiating events cornerstone objective for not limiting the likelihood of events that upset plant stability, specifically the cornerstone attribute of equipment performance.
f. Because this issue was entered into your corrective action program we identified this as a NCV of 10 CFR appendix B Criterion XVI "corrective actions" and Technical specification 3.4.13 "primary Coolant System Operational Leakage" for failure to prevent recurrence of leakage in CRDM housings due to TGSCC resulting in the operation of the reactor with pressure boundary leakage for greater than the TS allowed time.
g. The performance deficiency screened as green after screening under the initiating events cornerstone because we answered no to the question if after reasonable assessment of degradation, could the finding result in exceeding the RCS leak rate for a small LOCA and could the finding have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. Basically because of the nature of the cracking and your site procedures we believe it would be detected and corrected prior to reaching the small break LOCA limits.
h. We are not recommending a cross cutting aspect be applied to this performance deficiency though we do believe this issue is still indicative of current performance we are addressing this aspect in the next violation we will discuss.

3. Violation of root cause procedure
   a. When reviewing the 2012 root cause procedure related to the cracking identified in CRDM 24 we identified a failure to appropriately consider the generic implications of the cracking in the extent of condition review. The proposed corrective actions narrowly focused on the weld build up region instead of broader actions to ensure other CRDM housing welds were fit for their intended service life.
   b. You provided additional information to us to justify excluding these welds from the scope of the corrective actions.
   c. You credited the actions taken in 2001 and stated that these actions would produce compressive stresses on the ID of welds 3 and 4 making them immune to cracking. These actions included performing heat sink welding,, changing th design around weld 3 and specifying a smoother surface finish.
   d. We acknowledge that these actions would reduce the tensile stress at the ID surface and thus reduce the probability of initiating TGSCC
   e. However the information provided did not demonstrate that TGSCC would not occur because it did not demonstrate that tensile stress would be eliminated at the ID surface during operation. In particular repairs completed at the inner surface of weld 4 would result in high residual tensile stress at the inside surface of the weld which would promote the initiation of TGSCC.
   f. Repairs were also performed on weld 3 from the outer diameter surface of the weld and the assumption has been made that heat sink welding would be sufficient to ensure residual compressive stress would remain at the ID surface of weld 3 even

with repairs to the OD surface. However a detailed residual stress test or modeling has not been performed to confirm this assumption.

g. We identified that the three factors required for TGSCC could still be present at welds 3 and 4

   - Corrosive environment – Weld 3 would operate in a similar environment as the weld build up region of the CRDM housing. Weld No. 4 would be exposed to a lower operating temperature then the weld build up region, however TGSCC can still occur at 250 degrees Fahrenheit as evidenced by the previous operating experience with cracking identified in the seal housings that operate at even lower temperatures.

   - Susceptible material – Welds 3, 4 and 5 are composed of the same weld filler and base metal materials as the weld buildup region (e.g. weld filler material consistent with the type 316 stainless housing base metal). This material would be equally susceptible to TGSCC, as the type 347 stainless steel and weld filler materials used in the pre-2001 CRDM housing design that developed a through wall leak caused by TGSCC at weld No.3.

   - Tensile stresses - While it is assumed that the corrective actions taken in response to the 2001 leak will reduce the potential for tensile stresses to exist on the inner surface of CRDM housings at welds 3 and 4, especially in light of the repairs made to welds 3 and 4, it has not been conclusively demonstrated that these tensile stresses have been eliminated. As such it is not reasonable to conclude that tensile stresses are not present and, therefore, the potential for transgranular stress corrosion cracking has been eliminated.

h. The discussion of sensitization is not germane to the observed cracking. Sensitization is not required for transgranular stress corrosion cracking. The use of materials resistant to sensitization do not reduce the likelihood of transgranular stress corrosion cracking.

i. Despite the fact that the root cause for the leak in CRDM housing 24 indicts manufacturing issues and alignment, it also includes an unidentified stress. This stress, if it exists, may be present to a greater or lesser extent in other housings. Based on this, it is not clear that the absence of cracking in welds 3 and 4 of CRDM housing 24 is definitive evidence that welds 3 and 4 are not subject to cracking in other CRDM housings. It also should be noted that CRDM housing 24 is not listed as having undergone weld repairs to weld 4. Cracking at weld 4 is more likely in CRDM housings other than CRDM 24.Therefore we do not believe you have established a sufficient basis in the RCR to exclude welds 3 and 4 from the extent of condition review

j. We determined that the failure to adequately evaluate and document the generic implications of the cause of cracking identified in CRDM #23 in accordance with the root cause procedure EN-LI-118 was a performance deficiency.

k. We determined this issue was more than minor because it adversely affected the Initiating event cornerstone attribute of equipment performance and we answered yes to the question if left uncorrected would the performance deficiency have the potential to lead to a more significant safety concern.

l. Specifically, absent NRC identification, you would not have completed further evaluations or inspections of CRDM housing welds which could have resulted in additional CRDM housing failure and leakage by TGSCC
m. The performance deficiency screened as green after screening under the initiating events cornerstone because we answered no to the question if after reasonable assessment of degradation, could the finding result in exceeding the RCS leak rate for a small LOCA and could the finding have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. Basically because of the nature of the cracking and your site procedures we believe it would be detected and corrected prior to reaching the small break LOCA limits.
n. We are recommending a Cross cutting aspect in the area of human performance Decision Making, because conservative assumptions were not used in decision making. Specifically, did not use conservative assumptions when excluding welds 3 and 4 as being susceptible to TGSCC and therefore include them in the generic implications section of the root cause report. (Item H.1(b))
o. Because you entered this issue into your corrective action program we are characterizing this issue as a NCV of 10 CFR Appendix B Criterion V "Instructions, Procedures and Drawings", having a very low safety significance (Green), for failure to adequately evaluate the generic implications of the cause of cracking identified in CRDM #24 as it relates to weld 3 and 4 in accordance with the root cause procedure.

p. Title 10 CFR Appendix B Criterion V "Instruction, Procedures and Drawings requires in part, "Activities affecting quality shall be prescribed by documented procedures and shall be accomplished in accordance with these procedures."

q. Procedure EN-LI-118 Root Cause evaluation process revision 17 states:

    a. 5.5 (12)e: perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affects other SSC's , organizations or work processes. Use the two step process in accordance with attachment 9.7

    b. Attachment 9.7 states Determine whether the occurrence/consequence (problem) is isolated, or whether it has broader (generic or common mode) implications. Achieve this by asking the following questions:

        i. Could this happen to equipment that is similar in function, design, or service condition?

        ii. Could this happen to a group of components? (components of the same construction or materials that could be similarly affected by one condition)

    c. Attachment 9.7 also states: Document the results of the above considerations. Include the following items in the write up:

i. Generic Implications (Is this problem/ cause limited to this component/equipment, or does it apply to others as well)

ii. Existing broader (generic/common mode) considerations

d. 5.5(15)(10)c&f: Document proposed corrective actions and due dates to address valid generic implications. If no corrective action is recommended for a valid generic implication then document the basis for this conclusion and any risk or consequence identified as a result of taking no action.

r. Contrary to the above, you failed to perform an activity affecting quality in accordance with procedure EN-LI-118. Specifically, you did not identify and document the existing broader (generic/common mode) considerations associated with TGSCC at CRDM housing welds No. 3 and No. 4. Consequently, you failed to propose corrective actions for the generic implications of TGSCC at CRDM housing welds No. 3 and No. 4.

Questions?

In order for us to follow our process I wanted to take this opportunity to ask if you are planning on contesting any of these violations.

Thanks You

b. Findings

.1 Failure to Prevent Recurrence of a Significant Issue Adverse to Quality

Introduction: The inspectors identified a Green Non-Cited Violation (NCV) of 10 CFR Part 50 Appendix B Criterion XVI, and Technical Specification (TS) 3.4.14 Primary Coolant System (PCS) Operational Leakage, for failure to prevent recurrence of a significant condition adverse to quality resulting in a non-compliance with the TS. Specifically, the licensee failed to include the internal CRDM weld build-up area within the scope of corrective actions taken for a 2001 CRDM housing leakage event (a significant condition adverse to quality) and consequently leakage recurred at the CRDM housing No. 24 in 2012.

Description: On August 12, 2012 Palisades Nuclear Power Station shutdown to investigate an increase in unidentified leakage. During a walk-down performed post shutdown the licensee discovered the source of the leakage to be a pressure boundary leak from (CRDM) Housing No. 24. After further testing, the licensee determined the leak occurred because of a through-wall flaw adjacent to a weld build up on the interior of the housing (weld 5). Weld 5 consists of a weld material deposit applied to the inside diameter of the CRDM housing which provides for alignment of the CRDM.

The licensee formed a root cause team (RCT) staffed with licensee personnel and augmented with input from vendors. The root cause investigation was conducted in accordance with site procedure EN-LI-118 "Root Cause Evaluation Process" and was documented in root cause analysis report CR-PLP-2012-05623. In this report, the licensee's RCT determined that the probable cause of the cracking was:

> "Stresses in the weld build up area due to manufacturing irregularities and misalignments between CRD-24 upper housing, support tube, and the associated reactor head penetration/CRDM nozzle. Based on lack of cracking found in the other 8 upper housings tested, the failed CRD-24 upper housing contains an as-yet unidentified additional stress".

The RCT also identified the following contributing cause:

> "Transgranular Stress Corrosion Cracking (TGSCC) initiating within the internal weld build-up material of CRD-24. The through wall crack initiated in the weld material and then propagated through the base metal until a leak developed in the OD witness band region at the base of the inner diameter (ID) weld build up.

This conclusion was based upon destructive and non destructive examinations (NDE) completed on a section of the failed housing which included the through-wall flaw. The RCT also relied upon vendor technical reports assessing the results of the NDE as well as vendor calculations related to the stresses in the CRDM housings.

To determine the extent of condition, the licensee performed ultrasonic (UT) examinations of the weld build up area on 8 additional CRDM housings. The licensee selected these locations based on being in a similar location on the head as CRDM-24, and previous cracking having been identified in some of these locations prior to the replacement of the CRDM upper housings and seal housings. Additionally, the licensee was planning to conduct examinations of additional housings during the next refueling outage. The inspectors concluded that this was an appropriate initial extent of condition review based upon the cause of the CRDM No. 24 failure identified by the licensee.

In 2001, the licensee discovered a steam leak in the housing of CRDM-21 caused by a through-wall TGSCC at CRDM housing weld No. 3 which was located just below the weld build-up region. This issue was categorized as a significant issue adverse to quality (SCAQ) by the licensee (CPAL0102186) and the licensee's root cause evaluation was documented in RCR/C-PAL-01-02186. The licensee considered this issue a SCAQ because it met their procedure EN-LI-102 "Corrective Action Process" definition which stated the following definition for significant condition adverse to quality: "Conditions such as failures, malfunctions, deficiencies, deviations, defective material & equipment, and non-conformances which have resulted in, or could result in, a significant degradation or challenge to nuclear safety. The licensee concluded that the cracks in CRDM-21 were caused by TGSCC which occurred in areas of heavy grinding or machining tool marks. Specifically, this leak was the result of an inner diameter initiated, axially oriented, transgranular crack in the austenitic stainless steel housing material. The failure analysis performed in response to this event identified both axial and circumferential cracks associated with weld 3. Extent of condition inspections revealed additional, non-through wall cracks associated with weld 3 in 41 of the 44 remaining housings for a total of 42 of 45 housings containing cracks.

In response to the 2001cracking, Palisades replaced all 45 CRDM housings with housings thought to be more resistant to cracking. Principle changes included:

a. Elimination of weld number 2,

b. Relocation of weld number 3 to a higher location thereby minimizing the deposition of crud in the gap between the weld and the bottom plate of the rack and pinion assembly,

c. Reduction in residual stresses and cold work on welds by requiring better surface finishes, and

d. Use of heat sink welding to reduce ID residual tensile stresses.

Licensee corrective actions taken in response to the 2001 event were limited to pressure boundary welds and did not include the inspectors reviewed the licensee actions to determine if they had been sufficient to eliminate one of the 3 necessary factors to cause TGSCC on the CRDM housings: (1) a susceptible material, (2) a corrosive environment and (3) tensile stress." The inspectors identified that the licensee had failed to eliminate one or more of the necessary factors at the weld build-up area to preclude TGSCC in the replacement housing. Specifically:

- o The licensee's 2001 root cause report documented that the weld build-up region is exposed to essentially the same environment as the weld that experienced the cracking (corrosive environment remained unchanged).

- o No analysis was completed on the stress conditions for the weld build-up region prior to approving the modified replacement housing design (left residual tensile weld stresses on ID of CRDM surface).

- Fabrication restrictions to prohibit grinding were not applied to the weld build-up region (grinding promotes residual tensile stress state on ID of CRDM surface)

- Machining was performed on the weld build-up areas during the fabrication process in order to achieve the dimensions and geometry specified in the design. This process induced cold work stresses in the weld.

- Material was changed from type 347 to type 316 stainless steel (both materials are essentially equally susceptible to TGSCC).

Based upon the recurrence of through-wall leakage in the CRDM housings that occurred at the weld buildup region of the CRDM housings by TGSCC, the inspectors concluded that the licensee actions were not adequate because the appropriate actions to preclude recurrence were within the licensee's ability to foresee and implement. In 1991, the Fort Calhoun plant had experienced through-wall leakage due to TGSCC at the weld build-up region of their CRDM housings (same housing design) and this operational experience had been reviewed by the licensee and dismissed. In the licensee's 2001 root cause evaluation, the licensee reviewed the weld build-up region failure by TGSCC at Fort Calhoun in the spare housing and concluded it would not occur at Palisades. This conclusion was based on the assumption that a higher oxygen environment (more aggressive environment) would exist in the spare Fort Calhoun housings than in the inservice Palisades housings. However the licensee did not confirm this assumption, nor did the licensee perform additional testing to determine if the environment of their inservice housings was sufficiently benign to prevent TGSCC. The licensee's 2012 RCT reached a similar conclusion and documented that due to organizational/ programmatic weakness at Palisades, the 1991 Fort Calhoun operating experience was not adequately utilized to include inspection of the housing ID weld build-up regions. The inspectors identified that the licensee had missed a key opportunity to implement effective corrective actions that could have prevented recurrence of the 2001 leakage event and elected not to pursue these actions because of the cost. Specifically, in EA-EAR-2001-0426-01 the licensee considered fabricating the replacement housings with Inconel 600 material because it was much more resistant to TGSCC. However, the licensee elected not to fabricate the replacement housings using this material because of the increased cost.

In January of 2002, an NRC special inspection team (SIT) (reference IR 50-2555/01-15) reviewed the licensee proposed corrective actions associated with the through-wall leakage of the CRDM-21 housing caused by TGSCC. The 2001 root cause report reviewed by the NRC stated the action to prevent recurrence was to "develop and implement an inspection plan to address areas and components identified in Attachment C-Extent of Condition". One of the components included in Attachment C was the CRD Mechanism. The recommended action was to perform volumetric inspection of the welds contained in the CRD Mechanism. The table also refers to a susceptibility analysis (EA-C-PAL-01-2186-02 "CRD Upper Housing and Nozzle Weld Susceptibility Comparison" to identify how degradation can be identified in this component. The objective of this document was to provide justification as to why the first weld (weld 1) above the reactor head is deemed to be less susceptible than the upper housing welds to failure by TGSCC and should not be included in the extent of condition. The susceptibility analysis excludes weld 5 because it is a weld overlay and not a butt weld and was deemed to be less susceptible to TGSCC than the butt welds. By not including weld 5 in the susceptibility analysis the licensee did not evaluate the stresses, material and environment of this weld to conclude it is not susceptible to TGSCC. An attachment to this analysis states machining marks were present on weld 5 which was identified as a key contributor to the cracking identified in weld 3. After this analysis was complete the licensee decided to replace all CRDM housings with the new design and control the fabrication process on the butt welds and the inspection plan would consist of the required ASME inspections. Weld 5 was excluded from these corrective actions and no fabrication controls were placed on weld 5 to reduce the stresses in this location. Therefore, the inspectors concluded that the licensee did not effectively implement corrective actions for the 2001 CRDM housing leak resulting in the 2012 CRDM-24 housing leak.

During the 2012 NRC special inspection the NRC identified an unresolved item for the Technical Specification pressure boundary leak. The licensee determined the CRDM-24 leakage commenced on July 14, 2012-and that the plant continued to operate in this condition for greater than 6 hours, which ~~is~~ was greater than the required shutdown time with pressure boundary leakage per TS LCO 3.4.14. Based on the review discussed above, unresolved items 05000255/2012012-01 "TS for PCS Pressure Boundary Leakage" and 05000255/2012012-03 "Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality," are closed.

Analysis: The inspectors determined that the licensee's failure to prevent recurrence of TGSCC of the CRDM housings (a significance condition adverse to quality) that resulted in a TS non-compliance was a performance deficiency that warranted a significance evaluation. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. Specifically the licensee did not limit the likelihood of events that upset plant stability by not taking adequate corrective actions to prevent recurrence of leakage in CRDM housings which represents a pressure boundary leakage and a condition prohibited by the Technical Specifications. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a ~~large~~ component rupture.

The inspectors determined this finding was caused by the same errors that led to the violation discussed in section 4OA2 (b.2) of this report and is indicative of current performance. Because the very similar cause for this performance deficiency and the one discussed in Section 4OA2 (b.2) of this report, no separate cross-cutting aspect is assigned to this finding.

Enforcement: The inspectors identified a NCV of 10 CFR Appendix B Criterion XVI "Corrective Actions", and Technical Specification 3.4.13 "Primary Coolant System Operational Leakage", having a very low safety significance (Green), for failure to prevent the recurrence of leakage in CRDM housings due to TGSCC resulting in the operation of the reactor with pressure boundary leakage for greater than the TS allowed time.

10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that, for significant conditions adverse to quality, the cause of the condition is determined and corrective action taken to preclude repetition.

Contrary to the above, August 12, 2012, the licensee failed to take corrective action to preclude repetition for a significant condition adverse to quality. Specifically, June 21, 2001 the licensee discovered a through wall leak in CRDM 21 due to TGSCC and failed to include weld 5 in the corrective actions as discussed in the above description which resulted in a through wall leak in CRDM 24. The pressure boundary leakage at CRDM began on July 14, 2012 and the plant continued to operate until August 12, 2012 which exceeded the 6 hours allowed by TS 3.4.13.

The licensee took corrective actions related to the results of the current root cause report which included the development of an inspection plan that would inspect weld 5 every outage until all CRDM housing were inspected.

Because this violation was of very low safety significance and it was entered into the licensee's corrective action program as CR-PLP-2013-01134, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000255/2013002-*XX*; Failure to Prevent Recurrence of a Significant Condition Adverse to Quality).

Based on the information provided, we recognize the corrective actions applied to welds 3 and 4 in 2001 served to reduce the tensile stresses in the welds and, though not documented in the generic implications section, the root cause report does address to an extent the justification for not considering welds 3 and 4 susceptible to TGSCC.

However, the information does not demonstrate that compressive stresses will be retained at the inside surface. Specifically, the inside diameter repairs conducted at weld No. 4 locations would result in residual tensile stress at the inside surface which would promote the initiation of TGSCC. In particular, we noted that each of three factors required for TGSCC would still be present at welds 3 and 4:

- Corrosive environment
  - Though at a lower temperature than weld #5, TGSCC could still occur at 250 degrees Fahrenheit as evidenced by the seal housing cracks, therefore a lower temperature will not preclude TGSCC
- Susceptible material
  - Same material as Weld #5 and for TGSCC purposes, essentially the same material as welds #3, pre-2001
- Tensile stresses above a certain unknown level
  - Fabrication and repairs will result in tensile stress at the inside surface of weld 4 and possibly weld No. 3. The specific threshold for ~~of~~ tensile stress that needs to be exceeded is unknown and stresses in the weld are also unknown, especially those subject to grinding and re-weld.

The licensee also pointed out one of the reasons Weld #4 repairs are not an issue is because since the welding heat input is low and the carbon content of the material is low, then sensitization is not likely to occur. We agree that sensitization can be an issue for IGSCC, but we disagree that it is relevant to the TGSCC concern, since sensitization is not required for TGSCC to occur as evidenced by your metallurgical investigations into the root cause of the 2001 and 2011 housing failures.

For weld #3 with regards to the statement : "For all the lower flange to pipe structure welds (weld 3), the welds were either cut out or excavated from the OD and then replaced in accordance with the approved weld procedure which would preserve the advantages of LPHSW" , it is our position that an analysis of the actual weld and associated repair is necessary to determine the stress fields at the intersection of the new and existing welds in order to conclude the ID of the weld is unaffected by this process. We also noted the repairs were not performed using the LPHSW process unless the pressure boundary was broken, as documented in the repair traveler.

Another item of note is the root cause was identified as "stresses in the weld buildup area due to manufacturing irregularities and misalignments between CRD-24 upper housing, support tube, seismic supports and the associated reactor head penetration/CRDM nozzle." It also states that "based on the lack of cracking found in the other 8 upper housings tested, the failed CRD-24 upper housing contains an as-yet unidentified additional stress". Based on the potential existence of an additional stress, it cannot be concluded that this stress exists in only weld 5 of

CRDM housing 24. Therefore we do not agree that welds 3 and 4 should not have been considered when evaluating for generic implications. Furthermore we do not agree the justification provided is sufficient to exclude welds 3 and 4 from the inspection plan absent additional analysis specific to the welds at Palisades (considering operating stresses and stresses induced during the weld repair process)

In conclusion after reviewing the material provided, it does not offer evidence to exclude any of the three key elements needed to prevent TGSCC- the environment (same as prior TGSCC events), residual stress (may have a case for weld #3 but would need to provide analysis to confirm at NOP compressive ID stress, but they cannot make this case for weld #4 areas that have been repaired) and susceptible material (type 316SS)

Recommendation:

Violation of 10 CFR Appendix B Criterion V: Instructions Procedures and Drawings

- The performance deficiency being the failure to adequately consider welds 3 and 4 in the generic implications portion of the root cause report and therefore provide adequate justification for why no additional corrective actions associated with these welds are needed or take corrective action as necessary to address the potential generic implication.
- Specifically, our concern is that the licensee is applying a narrow focus for what the extent of the root and contributing causes may be, by essentially looking at only like for like scenarios rather than considering what other portions of the components could be subject to the same or similar factors and whether the factors that discount them are valid based on research, analysis, review and or inspection.
- We consider this a violation of 10 CFR Appendix B Criterion V, for failure to follow procedures. Specifically procedure EN-LI-118 Root Cause evaluation process states:
    a. 5.5 (12)e: perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affects other SSC's , organizations or work processes. Use the two step process in accordance with attachment 9.7
    b. Attachment 9.7 states Determine whether the occurrence/consequence (problem) is isolated, or whether it has broader (generic or common mode) implications. Achieve this by asking the following questions:
        i. Could this happen to equipment that is similar in function, design, or service condition?
        ii. Could this happen to a group of components? (components of the same construction or materials that could be similarly affected by one condition)
    c. Attachment 9.7 also states: Document the results of the above considerations. Include the following items in the write up:
        i. Generic Implications (Is this problem/ cause limited to this component/equipment, or does it apply to others as well)
        ii. Existing broader (generic/common mode) considerations

d. 5.5(15)(10)c&f: Document proposed corrective actions and due dates to address valid generic implications. If no corrective action is recommended for a valid generic implication then document the basis for this conclusion and any risk or consequence identified as a result of taking no action.

- Contrary to this the licensee did not fully consider the generic implication by not adequately evaluating the root and contributing causes for their affects on welds 3 and 4 and therefore did not develop corrective actions to address these or provided a basis for not taking any corrective actions related to the welds.
- We categorized this issue as more than minor because if left uncorrected it has the potential to lead to a more significant safety concern Specifically if you don't have adequate justification for not including welds 3 and 4 in your generic implications section and/or have adequate justification for not taking corrective actions to address the potential generic concern it may result in another through wall leak.
- The performance deficiency screened as green after screening under the initiating events cornerstone because we answered no to the question if after reasonable assessment of degradation, could the finding result in exceeding the RCS leak rate for a small LOCA and could the finding have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. Basically because of the nature of the cracking and your site procedures we believe it would be detected and corrected prior to reaching the small break LOCA limits.
- We are proposing a cross cutting aspect in the area of Human Performance, decision making for non-conservative assumptions. Specifically we believe you did not use conservative assumptions when electing not to include welds 3 and 4 for potential generic implication.

REPORT DETAILS

4. REACTOR SAFETY

4OA2 Identification and Resolution of Problems (71152)

.3 Selected Issue Follow-up Inspection: Through Wall Leakage of Control Rod Drive Mechanism (CRDM) Housing #24

a. Inspection Scope

On August 12, 2012 the licensee shutdown to investigate an increase in unidentified leakage. The source of the leakage was determined to be a crack in control rod drive mechanism housing (CRDM) No. 24. Shortly after the discovery of the leak in CRDM housing No. 24, the NRC dispatched a special inspection team (SIT) to review the CRDM No. 24 leakage event. The SIT identified an unresolved item (URI) related to the potential failure to prevent recurrence of a significant condition adverse to quality (SCAQ) which was considered an unresolved item, because the licensee's root cause investigation was ongoing at that time. The licensee subsequently removed the failed housing from service for further testing and completed an evaluation to determine the cause of the cracking (CR-PLP-2012-05623).

From March 4, 2013 to March 15, 2013, the inspectors completed one inspection sample regarding problem identification and resolution based upon review of the licensee's root cause report contained in corrective action document CR-PLP-2012-05623.

The inspectors reviewed the licensee's actions in accordance with performance attributes identified in IP 71152. Specifically, the inspectors reviewed licensee corrective action records to determine if: (1) the problems were accurately identified; (2) operability and reportability were adequately ascertained; (3) extent of condition and generic implications were appropriately addressed; (4) classification and prioritization of problem was commensurate with safety significance; (5) root and contributing causes were identified; (6) corrective actions were appropriately focused to correct problem; and (7) timely corrective actions were completed or proposed commensurate with the safety significance of the issues.

b. Findings

.1 Failure to Prevent Recurrence of a Significant Issue Adverse to Quality

Introduction: The inspectors identified a Green Non-Cited Violation (NCV) of 10 CFR Part 50 Appendix B Criterion XVI, and Technical Specification (TS) 3.4.14 Primary Coolant System (PCS) Operational Leakage, for failure to prevent recurrence of a significant condition adverse to quality resulting in a non-compliance with the TS. Specifically, the licensee failed to include the internal CRDM weld build-up area within the scope of corrective actions taken for a 2001 CRDM housing leakage event (a significant condition adverse to quality) and consequently leakage recurred at the CRDM housing No. 24 in 2012.

Description: On August 12, 2012 Palisades Nuclear Power Station shutdown to investigate an increase in unidentified leakage. During a walk-down performed post

shutdown the licensee discovered the source of the leakage to be a pressure boundary leak from (CRDM) Housing No. 24. After further testing, the licensee determined the leak occurred because of a through-wall flaw adjacent to a weld build up on the interior of the housing (weld 5). Weld 5 consists of a weld material deposit applied to the inside diameter of the CRDM housing which provides for alignment of the CRDM.

The licensee formed a root cause team (RCT) staffed with licensee personnel and augmented with input from vendors. The root cause investigation was conducted in accordance with site procedure EN-LI-118 "Root Cause Evaluation Process" and was documented in root cause analysis report CR-PLP-2012-05623. In this report, the licensee's RCT determined that the probable cause of the cracking was:

> "Stresses in the weld build up area due to manufacturing irregularities and misalignments between CRD-24 upper housing, support tube, and the associated reactor head penetration/CRDM nozzle. Based on lack of cracking found in the other 8 upper housings tested, the failed CRD-24 upper housing contains an as-yet unidentified additional stress".

The RCT also identified the following contributing cause:

> "Transgranular Stress Corrosion Cracking (TGSCC) initiating within the internal weld build-up material of CRD-24. The through wall crack initiated in the weld material and then propagated through the base metal until a leak developed in the OD witness band region at the base of the inner diameter (ID) weld build up.

This conclusion was based upon destructive and non destructive examinations (NDE) completed on a section of the failed housing which included the through-wall flaw. The RCT also relied upon vendor technical reports assessing the results of the NDE as well as vendor calculations related to the stresses in the CRDM housings.

To determine the extent of condition, the licensee performed ultrasonic (UT) examinations of the weld build up area on 8 additional CRDM housings. The licensee selected these locations based on being in a similar location on the head as CRDM-24, and previous cracking having been identified in some of these locations prior to the replacement of the CRDM upper housings and seal housings. Additionally, the licensee was planning to conduct examinations of additional housings during the next refueling outage. The inspectors concluded that this was an appropriate initial extent of condition review based upon the cause of the CRDM No. 24 failure identified by the licensee.

In 2001, the licensee discovered a steam leak in the housing of CRDM-21 caused by a through-wall TGSCC at CRDM housing weld No. 3 which was located just below the weld build-up region. This issue was categorized as a significant issue adverse to quality (SCAQ) by the licensee (CPAL0102186) and the licensee's root cause evaluation was documented in RCR/C-PAL-01-02186. The licensee considered this issue a SCAQ because it met their procedure EN-LI-102 "Corrective Action Process" definition which stated the following definition for significant condition adverse to quality: "Conditions such as failures, malfunctions, deficiencies, deviations, defective material & equipment, and non-conformances which have resulted in, or could result in, a significant degradation or challenge to nuclear safety. The licensee concluded that the cracks in CRDM-21 were caused by TGSCC which occurred in areas of heavy grinding or machining tool marks. Specifically, this leak was the result of an inner diameter initiated, axially oriented, transgranular crack in the austenitic stainless steel housing material. The failure analysis performed in response to this event identified both axial and circumferential cracks associated with weld 3. Extent of condition inspections revealed additional, non-through wall cracks associated with weld 3 in 41 of the 44 remaining housings for a total of 42 of 45 housings containing cracks.

In response to the 2001cracking, Palisades replaced all 45 CRDM housings with housings thought to be more resistant to cracking. Principle changes included:

a. Elimination of weld number 2,

b. Relocation of weld number 3 to a higher location thereby minimizing the deposition of crud in the gap between the weld and the bottom plate of the rack and pinion assembly,

c. Reduction in residual stresses and cold work on welds by requiring better surface finishes, and

d. Use of heat sink welding to reduce ID residual tensile stresses.

Licensee corrective actions taken in response to the 2001 event were limited to pressure boundary welds and did not include the inspectors reviewed the licensee actions to determine if they had been sufficient to eliminate one of the 3 necessary factors to cause TGSCC on the CRDM housings: (1) a susceptible material, (2) a corrosive environment and (3) tensile stress." The inspectors identified that the licensee had failed to eliminate one or more of the necessary factors at the weld build-up area to preclude TGSCC in the replacement housing. Specifically:

- The licensee's 2001 root cause report documented that the weld build-up region is exposed to essentially the same environment as the weld that experienced the cracking (corrosive environment remained unchanged).

- No analysis was completed on the stress conditions for the weld build-up region prior to approving the modified replacement housing design (left residual tensile weld stresses on ID of CRDM surface).

- Fabrication restrictions to prohibit grinding were not applied to the weld build-up region (grinding promotes residual tensile stress state on ID of CRDM surface)

- Machining was performed on the weld build-up areas during the fabrication process in order to achieve the dimensions and geometry specified in the design. This process induced cold work stresses in the weld.

- Material was changed from type 347 to type 316 stainless steel (both materials are essentially equally susceptible to TGSCC).

Based upon the recurrence of through-wall leakage in the CRDM housings that occurred at the weld buildup region of the CRDM housings by TGSCC, the inspectors concluded that the licensee actions were not adequate because the appropriate actions to preclude recurrence were within the licensee's ability to foresee and implement. In 1991, the Fort Calhoun plant had experienced through-wall leakage due to TGSCC at the weld build-up

region of their CRDM housings (same housing design) and this operational experience had been reviewed by the licensee and dismissed. In the licensee's 2001 root cause evaluation, the licensee reviewed the weld build-up region failure by TGSCC at Fort Calhoun in the spare housing and concluded it would not occur at Palisades. This conclusion was based on the assumption that a higher oxygen environment (more aggressive environment) would exist in the spare Fort Calhoun housings than in the inservice Palisades housings. However the licensee did not confirm this assumption, nor did the licensee perform additional testing to determine if the environment of their inservice housings was sufficiently benign to prevent TGSCC. The licensee's 2012 RCT reached a similar conclusion and documented that due to organizational/ programmatic weakness at Palisades, the 1991 Fort Calhoun operating experience was not adequately utilized to include inspection of the housing ID weld build-up regions. The inspectors identified that the licensee had missed a key opportunity to implement effective corrective actions that could have prevented recurrence of the 2001 leakage event and elected not to pursue these actions because of the cost. Specifically, in EA-EAR-2001-0426-01 the licensee considered fabricating the replacement housings with Inconel 600 material because it was much more resistant to TGSCC. However, the licensee elected not to fabricate the replacement housings using this material because of the increased cost.

In January of 2002, an NRC special inspection team (SIT) (reference IR 50-2555/01-15) reviewed the licensee proposed corrective actions associated with the through-wall leakage of the CRDM-21 housing caused by TGSCC. The 2001 root cause report reviewed by the NRC stated the action to prevent recurrence was to "develop and implement an inspection plan to address areas and components identified in Attachment C-Extent of Condition". One of the components included in Attachment C was the CRD Mechanism. The recommended action was to perform volumetric inspection of the welds contained in the CRD Mechanism. The table also refers to a susceptibility analysis (EA-C-PAL-01-2186-02 "CRD Upper Housing and Nozzle Weld Susceptibility Comparison" to identify how degradation can be identified in this component. The objective of this document was to provide justification as to why the first weld (weld 1) above the reactor head is deemed to be less susceptible than the upper housing welds to failure by TGSCC and should not be included in the extent of condition. The susceptibility analysis excludes weld 5 because it is a weld overlay and not a butt weld and was deemed to be less susceptible to TGSCC than the butt welds. By not including weld 5 in the susceptibility analysis the licensee did not evaluate the stresses, material and environment of this weld to conclude it is not susceptible to TGSCC. An attachment to this analysis states machining marks were present on weld 5 which was identified as a key contributor to the cracking identified in weld 3. After this analysis was complete the licensee decided to replace all CRDM housings with the new design and control the fabrication process on the butt welds and the inspection plan would consist of the required ASME inspections. Weld 5 was excluded from these corrective actions and no fabrication controls were placed on weld 5 to reduce the stresses in this location. Therefore, the inspectors concluded that the licensee did not effectively implement corrective actions for the 2001 CRDM housing leak resulting in the 2012 CRDM-24 housing leak.

→ Add additional opportunities existed to identify weld #5 as susceptible to TGSCC.

During the 2012 NRC special inspection the NRC identified an unresolved item for the Technical Specification pressure boundary leak. The licensee determined the CRDM-24 leakage commenced on July 14, 2012-and that the plant continued to operate in this condition for greater than 6 hours, which ~~is~~ was greater than the required shutdown time with pressure boundary leakage per TS LCO 3.4.14. Based on the review discussed above, unresolved items 05000255/2012012-01 "TS for PCS Pressure Boundary Leakage" and 05000255/2012012-03 "Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality," are closed.

Analysis: The inspectors determined that the licensee's failure to prevent recurrence of TGSCC of the CRDM housings (a significance condition adverse to quality) that resulted in a TS non-compliance was a performance deficiency that warranted a significance evaluation. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. Specifically the licensee did not limit the likelihood of events that upset plant stability by not taking adequate corrective actions to prevent recurrence of leakage in CRDM housings which represents a pressure boundary leakage and a condition prohibited by the Technical Specifications. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a ~~large~~ component rupture.

The inspectors determined this finding was caused by the same errors that led to the violation discussed in section 4OA2 (b.2) of this report and is indicative of current performance. Because the very similar cause for this performance deficiency and the one discussed in Section 4OA2 (b.2) of this report, no separate cross-cutting aspect is assigned to this finding.

Enforcement: The inspectors identified a NCV of 10 CFR Appendix B Criterion XVI "Corrective Actions", and Technical Specification 3.4.13 "Primary Coolant System Operational Leakage", having a very low safety significance (Green), for failure to prevent the recurrence of leakage in CRDM housings due to TGSCC resulting in the operation of the reactor with pressure boundary leakage for greater than the TS allowed time.

10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that, for significant conditions adverse to quality, the cause of the condition is determined and corrective action taken to preclude repetition.

Contrary to the above, August 12, 2012, the licensee failed to take corrective action to preclude repetition for a significant condition adverse to quality. Specifically, June 21, 2001 the licensee discovered a through wall leak in CRDM 21 due to TGSCC and failed to include weld 5 in the corrective actions as discussed in the above description which resulted in a through wall leak in CRDM 24. The pressure boundary leakage at CRDM began on July 14, 2012 and the plant continued to operate until August 12, 2012 which exceeded the 6 hours allowed by TS 3.4.13.

The licensee took corrective actions related to the results of the current root cause report which included the development of an inspection plan that would inspect weld 5 every outage until all CRDM housing were inspected.

Because this violation was of very low safety significance and it was entered into the licensee's corrective action program as CR-PLP-2013-01134, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000255/2013002-*xx*; Failure to Prevent Recurrence of a Significant Condition Adverse to Quality).

.2 <u>Failure to Adequately Address the Generic Implications of the Cracking identified in CRDM 24</u>

<u>Introduction:</u> The inspectors identified a Green Non-Cited Violation (NCV) of 10 CFR Part 50 Appendix B Criterion V, for failure to follow the root cause procedure. Specifically, the licensee failed to adequately evaluate the generic implications of the cause of the cracking identified in CRDM No. 24.

<u>Description:</u> While reviewing the 2012 root cause report CR-PLP-2013-05623 related to the cracking identified in CRDM No. 24 the inspectors identified that the licensee had not appropriately considered the generic implications of the cracking in the extent of condition review. The licensee's proposed corrective actions narrowly focused on the weld build up region (weld 5), instead of broader actions to ensure other CRDM housing welds were fit for their intended service life.

On March 13, the inspectors requested that the licensee identify the bases for excluding other CRDM housing welds (weld #3 below the weld build up region and weld #4 above the weld build up region) from the scope of planned corrective actions. On March 29, the licensee provided additional information to justify excluding these welds from the scope of the corrective actions. The licensee credited the corrective actions associated with the modifications to the CRDM housing design completed in 2001 as the basis to exclude housing welds No 3 and 4 from additional actions to identify the extent of TGSCC. The corrective actions taken in 2001 included performing heat sink welding, which is a methodology used to reduce the stresses on the inner diameter (ID) of the weld, they also changed the design to reduce design stresses at weld #3 and they specified a smoother surface finish (RMS 125) to reduce potential crack initiation points. The licensee stated that these actions would produce compressive stresses on the ID of welds 3 and 4 making them immune from cracking. The inspectors acknowledged that these actions would reduce the tensile stress at the ID surface and thus reduce the probability of initiating TGSCC.

However, the information provided did not demonstrate that TGSCC would not occur because it did not demonstrate that tensile stress would be eliminated at the ID surface during operation. In particular, repairs completed at the inner surface of weld No. 4, would result in high residual tensile stress at the inside surface of the weld which would promote the initiation of TGSCC. Repairs were also performed on weld No. 3; from the outer diameter (OD) surface of the weld. The licensee believed that the last pass heat sink welding process would be sufficient to ensure residual compressive stress would remain at the ID surface of Weld No. 3 even with repairs to the OD surface. However, the licensee had not completed detailed residual weld stress testing or modeling to confirm this assumption.

The inspectors identified that the three factors required for TGSCC could still be present at the welds 3 and 4 as follows:

- Corrosive environment – Weld 3 would operate in a similar environment as the weld build up region of the CRDM housing. Weld No. 4 would be exposed to a lower operating temperature then the weld build up region, however TGSCC can still occur at 250 degrees Fahrenheit as evidenced by the Palisades previous operating experience with cracking identified in the seal housings that operate at even lower temperatures.

- Susceptible material – Welds 3, 4 and 5 are composed of the same weld filler and base metal materials as the weld buildup region (e.g. weld filler material consistent with the type 316 stainless housing base metal). This material would be equally susceptible to TGSCC, as the type 347 stainless steel and weld filler materials used in the pre-2001 CRDM housing design that developed a through wall leak caused by TGSCC at weld No.3.

- Tensile stresses - While it is assumed that the corrective actions taken in response to the 2001 leak will reduce the potential for tensile stresses to exist on the inner surface of CRDM housings at welds 3 and 4, especially in light of the repairs made to welds 3 and 4, it has not been conclusively demonstrated that these tensile stresses have been eliminated. As such it is not reasonable to conclude that tensile stresses are not present and, therefore, the potential for transgranular stress corrosion cracking has been eliminated.

Although the root cause report discusses manufacturing irregularities and misalignment between CRDM housing 24 and the support tube, seismic supports and the associated reactor head penetration/CRDM nozzle as potential source of stresses leading to cracking, the root cause report also states that "based on the lack of cracking found in the other 8 upper housings tested, the failed CRD-24 upper housing contains an as-yet unidentified additional stress." Because the cause of the additional stress was not identified, the licensee had not established a sufficient basis in the RCR to exclude welds 3 and 4 from the extent of condition review (e.g. potential generic implications).

The inspectors identified that the licensee had not followed Procedure EN-LI-118 Root Cause evaluation in the root cause review of the CRDM housing No. 24 leak as documented in report CR-PLP-2013-05623. Section 5.5 (12)e of EN-LI-118 required that the licensee "perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affects other SSC's." In this case, the inspectors identified that the licensee had not documented a sufficient basis in RCR CR-PLP-2013-05623 to exclude welds No. 3 and No. 4 from the generic factors discussed above that led to the 2012 leak in the CRDM housing No. 24 (e.g. TGSCC at the weld buildup region). The licensee entered this issue into the corrective action program as CR-PLP-2013-01500. To restore compliance with the procedure, the licensee intended to revise the inspection plan to add additional corrective actions to inspect a sample of welds No. 3 and 4 for TGSCC during the upcoming refueling outage.

Analysis: The inspectors determined that the failure to adequately evaluate the generic implications of the cause of the cracking identified in CRDM #24 in accordance with the root cause procedure EN-LI-118 was a performance deficiency that warranted a significance evaluation. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. The inspectors also answered "yes" to the More-than-Minor screening questions "if left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, absent NRC identification, the licensee would not have completed further evaluations or inspections of CRDM housing welds which could have resulted in additional CRDM housing failure and leakage by TGSCC. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a large component rupture.

The inspectors determined that the primary cause of the failure to adequately consider welds 3 and 4 on the generic implications section of the root cause report related to the cross-cutting component of Human Performance, Decision Making, because licensee staff did not use conservative assumptions in decision making. Specifically, the licensee did not use conservative assumptions when excluding welds 3 and 4 as being susceptible to TGSCC and therefore include them in the generic implications section of the root cause report. (Item H.1(b) of IMC 310).

Enforcement: The inspectors identified a NCV of 10 CFR Appendix B Criterion V "Instructions, Procedures and Drawings", having a very low safety significance (Green), for failure to adequately evaluate the generic implications of the cause of cracking identified in CRDM #24 as it relates to weld 3 and 4 in accordance with the root cause procedure.

Title 10 CFR Appendix B Criterion V "Instruction, Procedures and Drawings requires in part, "Activities affecting quality shall be prescribed by documented procedures and shall be accomplished in accordance with these procedures."

Procedure EN-LI-118 Root Cause evaluation process revision 17 states:

a. 5.5 (12)e: perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affects other SSC's , organizations or work processes. Use the two step process in accordance with attachment 9.7
b. Attachment 9.7 states Determine whether the occurrence/consequence (problem) is isolated, or whether it has broader (generic or common mode) implications. Achieve this by asking the following questions:
    i. Could this happen to equipment that is similar in function, design, or service condition?
    ii. Could this happen to a group of components? (components of the same construction or materials that could be similarly affected by one condition)
c. Attachment 9.7 also states: Document the results of the above considerations. Include the following items in the write up:
    i. Generic Implications (Is this problem/ cause limited to this component/equipment, or does it apply to others as well)
    ii. Existing broader (generic/common mode) considerations
d. 5.5(15)(10)c&f: Document proposed corrective actions and due dates to address valid generic implications. If no corrective action is recommended for a valid generic implication then document the basis for this conclusion and any risk or consequence identified as a result of taking no action.

Contrary to the above, from February 24, 2013 through April 18, 2013, the licensee failed to perform an activity affecting quality in accordance with procedure EN-LI-118. Specifically, the licensee did not identify and document the existing broader (generic/common mode) considerations associated with TGSCC at CRDM housing welds No. 3 and No. 4. Consequently, the licensee failed to propose corrective actions for the generic implications of TGSCC at CRDM housing welds No. 3 and No. 4. The licensee was considering adding welds 3 and 4 into their inspection plan for activities to be performed during the next refueling outage. Because of the very low safety significance and because the licensee entered this issue into their corrective action program (CR-PLP-2013-01500), it is being treated as a NCV consistent with Section 2.3.2 of the Enforcement Policy (NCV 05000255/2013003-xx).

Sanchez Santiago, Elba

From: Phillips, Charles
Sent: Wednesday, August 29, 2012 10:02 AM
To: Shah, Swetha
Cc: Shaikh, Atif; Sanchez Santiago, Elba
Subject: FW: Palisades startup

From: Phillips, Charles
Sent: Wednesday, August 29, 2012 11:01 AM
To: Giessner, John
Subject: RE: Palisades startup

Jack,

I should have been more clear. We will be ready to exit assuming we get several documents from the licensee in a timely manner in order to get time to review them. However, if we have extended conversations with PA (**and I understand the necessity for that**) then we may not get the time necessary to perform the reviews and **we may not be ready to exit on Friday.**

Jack you had a question on enforcement:

Findings Summary

NRC Identified Findings	1
Licensee Identified Findings	0
Minor Findings	3
Unresolved Items	2

NRC identified finding on inadequate extent of condition due to lack of coverage during the original UT of the 8 EOC CRDMs

Unresolved item 1 – Pressure boundary leakage. The Davis Besse Report stated we gave them enforcement discretion because "the licensee appropriately implemented their quality control program, and this violation was the result of unavoidable equipment failure," - We won't know that until the root cause is complete.

Unresolved item 2 – Corrective actions to prevent recurrence from 2001 event – were they accurate, we won't know that until they complete the root cause.

Minor findings:

Failure to follow ONP 23.1
Failure to follow UT procedure when performing original 8 CRDM EOC UTs
Failure to have procedure available when setting up to perform UT of CRDM 40

C13

Hills, David

From: Taylor, Thomas
Sent: Thursday, March 07, 2013 12:13 PM
To: Sanchez Santiago, Elba; Alley, David; Lennartz, Jay; Shah, Swetha; Giessner, John; Scarbeary, April; Hills, David; Betancourt, Diana
Subject: FW: Palisades Week 1 status
Attachments: Palisades CRDM Inspection Week 1 Status Meeting.doc

For the call later

Tom Taylor
US NRC Senior Resident Inspector
Palisades Nuclear Plant
269-764-8971 (w)
Thomas.Taylor@nrc.gov

From: Sanchez Santiago, Elba
Sent: Thursday, March 07, 2013 12:57 PM
To: Taylor, Thomas
Subject: Palisades Week 1 status

C14

Palisades CRDM Inspection Week 1 Status Meeting

Inspection to address 3 URI's

1. Violation of tech specs for operating with pressure boundary leakage

2. Potential inadequate corrective actions taken in 2001 to prevent recurrence

3. Potential discrepancies with the licensee's calculation for crack growth rate

Status

Violation of tech specs for operating with pressure boundary leakage

1. Plant did operate with pressure boundary leakage

2. Enforcement discretion often given if plant can demonstrate that pressure boundary leakage could not reasonably be known or that it could not reasonably have been expected or prevented

3. In this case enforcement discretion may depend on whether NRC believes actions to prevent recurrence taken in 2001 are sufficient

4. Violation under consideration

5. Consideration of this issue will continue next week

Potential inadequate corrective actions taken in 2001 to prevent recurrence

1. Not currently clear to NRC that plant actions in 2001 were sufficient to prevent recurrence

2. Information available to plant appears to include

    a. Failure of weld 5 at Ft Calhoun in oxygenated, stagnant environment

    b. Failure of seal housings in lower oxygen vented environments

    c. Failure of weld 3 in lower oxygen vented environment

    d. No clear distinction in stress levels (cold work) among seal housing welds, weld 5, and weld 3

3. Based on information available to plant, it is not clear to NRC that sufficient justification existed in 2001 to not address the potential for cracking of weld 5

4. NRC considering violation of 10 CFR 50 App B criterion 16, prevention of recurrence

5. NRC questioning whether granting of enforcement discretion for operating with pressure boundary leakage is appropriate given that actions to prevent recurrence of 2001 event appear insufficient.

6. Consideration of this issue will continue next week

Potential discrepancies with the licensee's calculation for crack growth rate

1. Resolution of this URI involves concepts rather than procedures or specific documents

2. The NRC finds that the licensee has considered a number of possible crack growth scenarios and has selected an inspection interval which bounds all reasonable crack growth rates. In the process of reaching this conclusion the NRC reviewed a variety of documents related to the root cause and the inspection plan. The NRC agreed with many but not all aspects of these documents. The NRC identified a few areas in which modifications to the existing documents are necessary to assure consistency between documents and provide assurance that the inspection plan will continue to bound all reasonable crack growth rates as it is implemented. Resolution of the URI is dependent on resolution of these issues.

3. NRC opinions are based primarily on a review of:

    a. Current Root Cause

    b. 2001 Root Cause

    c. B&W Failure Analysis

    d. 2001 B&W Crack Growth Rate Analysis

    e. 2001 Presentation by Ft Calhoun to NRC regarding weld 5 crack

    f. Palisades procedure "Primary Coolant System – Cooldown"

4. In its review of the root cause statement (root cause p33) the NRC:

    a. Concurs with the conclusion that the CRDM housing leak was due to transgranular stress corrosion cracking

    b. Concurs with the concept that manufacturing anomalies in CRDM housing 24 are significant to the failure

    c. Acknowledges that finite element analyses have not been able to conclusively demonstrate that the manufacturing anomalies create sufficient stresses to produce the observed cracking pattern

    d. Based on the inability of the finite element analyses to demonstrate the basis for cracking, acknowledges the need for the concept of an "as yet unidentified additional stress"

e. Does not concur that the lack of cracking in the 8 CRDM housings examined as extent of condition necessarily confirms that those housings do not have sufficient stresses to cause cracking. The stresses in those housings may be lower than in CRDM housing 24 and the cracking process may still be in the incubation phase.

f. Does not find that sufficient evidence has been presented in the Root Cause to support the conclusion that CRDM housing 24 is unique among all the CRDM housings in relation to the development TGSCC

5. In its review of the "Crack Growth Evaluation" section of the 2012 Root Cause Report (p41) the NRC:

a. Notes that one of the B&W Laboratory Conclusions (2012 Root Cause p 32) is "... it could not be conclusively determined if the beach marks corresponded to refueling outages (i.e. 18 month cycle) or shorter periods as occurred during outages over the past 24 months.

b. Notes that paragraph 3 of the Crack Growth Evaluation section states that "... appear to align nicely with the number of fuel cycles at the plant since the housings were replace in 2001"

c. Notes that similar beach marks were observed in the weld 5 failure at Ft Calhoun (2001) and the weld 3 failure at Palisades

d. Notes that in both cases the beach marks were correlated to pressure cycles (rather than fuel cycles)

e. Notes that a crack growth calculation in paragraph 4 of the crack growth evaluation section is based on 11 months. The NRC proposes that this should be 24 months as some of the shutdowns at Palisades during this period were hot shutdowns which did not result in pressure or temperature cycles which would create a beach mark.

f. Notes that refueling cycles are not the only events which will introduce oxygen into the CRDM housings. Based on information contained Palisades procedure "Primary Coolant System Cooldown" air will be introduced into the housings when primary coolant pump seals are repaired and may be introduced when CRDM seals are replaced. At least one such event occurred in the 6 refueling cycles which occurred prior to the failure of CRDM housing 24

g. Based on the above observations it appears that the crack growth rates contained in the crack growth rate based on operating cycles should be increased from its present value. It also appears that the crack growth rate based on heat up and cool down cycles should be reduced from its present value. The new value for the heat up and cool down crack growth rate appears consistent with other crack growth rates mentioned in other events and should not, therefore, be characterized as "ultraconservative"

h. Notes that in this event, if beach marks are correlated the refueling outages, the crack initiation period is very short compared to the period of crack growth.

i. Notes that for this investigation as well as other investigations, when beach marks are correlated with heat up and cool down cycles, the period of crack initiation is, as expected, substantially longer than the period of crack growth

6. In its review of the "Inspection Frequency" section of the 2012 Inspection Plan (p7) the NRC:

   a. Notes that the inspection plan contains only a recommendation for implementation of the plan rather than language such as a commitment to implement the plan

   b. Notes that this paragraph of the inspection plan conclusively attributes the failure of CRDM housing 24 to a manufacturing defect, while the Root Cause Evaluation qualifies this finding

   c. Notes this paragraph of the inspection plan conclusively attributes this defect only to CRDM housing 24

   d. Notes that this paragraph of the inspection plan states that 4 to 5 operating cycles are required for a crack to grow through wall. While the NRC finds the inspection interval used in the plan, i.e., inspections conducted every refueling outage to bounds all reasonable crack growth rates, the NRC finds that this statement, 4-5 operating cycles required for a crack to grow through wall, to be both inconsistent with the Root Cause Evaluation and to be non conservative.

7. While the NRC finds that the inspection interval currently proposed bounds all reasonable crack growth rates, the NRC would have increased assurance that the inspection plan would be implemented in an acceptable manner if the following document sections were modified as discussed above:

   a. Root Cause (Root Cause Evaluation p33)

   b. Crack Growth Evaluation section (Root Cause Report p41)

   c. Inspection Frequency section (Inspection Plan p7)

Other

1. While not specifically part of any of the URIs being addressed in this inspection, the NRC notes that the inspection plan does not address:

   a. Ongoing inspections of CRDM housings after the 4 refueling outage period currently covered in the plan

b. Inspection of welds 3 and 4 in addition the inspections of weld 5 currently included in the inspection plan

2. The NRC believes that these issues merit consideration because:

   a. The stress state of CRDM housing 24 appears to have resulted in both a rapid initiation and growth of cracks at weld 5. The remaining housings, which may have lower stresses, may require a much longer initiation time prior to the development of an identifiable crack. This period may or may not extend beyond the period of the currently proposed inspection plan. While a fully detailed ongoing plan may not be required at this point, the NRC believes that he current plan should address, in some manner, the concept of ongoing inspections

   b. Based on the information currently contained in the Root Cause Evaluation, it is not clear to the NRC that welds 3 and 4 are sufficiently different from weld 5 so as to preclude the need for an inspection program similar to that currently proposed for weld 5.

Hills, David

From: Hills, David
Sent: Thursday, March 07, 2013 4:45 PM
To: Hills, David
Subject: Palisades Debrief 3/7/13

Likely will develop performance deficiency related to licensee actions to address previous CRDM housing leakage and hence will likely not offer discretion from T.S. leakage violation for recent CRDM housing leak. Also, licensee crack growth analysis says will take 4 years to grow. We believe insufficient basis to support that (depends on what you consider the beach marks (operating cycles or heatup/cooldown cycles)), but regardless of that conclusion, licensee plans sample each of the next four outages. However, doesn't mean can changed mind based on previous conclusion. Also, licensee plans eddy current exam of samples. Need to ensure that they plan to do a demonstration that will ensure can distinguish between cracks and surface scratches. (Otherwise in same boat as with PNNL evaluation of previous exam data.)

C15

RELEASE IN ENTIRETY

Giessner, John

From: Sanchez Santiago, Elba
Sent: Tuesday, March 12, 2013 10:06 AM
To: Taylor, Thomas
Cc: Scarbeary, April
Subject: FW: Palisades failure to prevent recurrence
Attachments: pros and cons -msh input.docx

FYI

The attached document summarizes my current approach on the prevent recurrence issue and includes some refuting arguments started by Mel. I'll be adding additional arguments as I get more information.

From: Holmberg, Mel
Sent: Tuesday, March 12, 2013 10:55 AM
To: Sanchez Santiago, Elba
Subject: RE: Palisades failure to prevent recurrence

Elba,

Good initiative and I think you're on the right track. Suggest you also refute each of the licensee points/arguments that I have started this for you in the attached input. Some of my input about content of industry OE is based on my assumptions so please confirm my statements or ask licensee to confirm them.

M

From: Sanchez Santiago, Elba
Sent: Tuesday, March 12, 2013 9:12 AM
To: Holmberg, Mel
Subject: Palisades failure to prevent recurrence

Mel,

Attached is the list I came up which refutes and/or supports the licensee's position that they made a reasonable decision based on the information they had at the time. Let me know when you are available to discuss.

Thanks,
Elba

C/D

Licensee's point:

- The Fort Calhoun incident was specific to their "spare" housings which were in a high oxygenated environment- **Necessary environmental conditions were not determined from this OE. In particular the minimum oxygen threshold to support TGSCC was not established so that susceptible environmental conditions should not have been ruled out for Palisades based upon subjective judgments.**
- After years of testing Fort Calhoun did not identify this conditions in any other of their weld #5s- **What testing was done on the other inservice housings at weld 5? Do they mean that they did not see failures so they assumed no cracks? If no specific followup inspections (at weld 5) for inservice housing was conducted this statement has no merit.**
- The Palisades weld #3 issue involved very high stresses due to the environment it was exposed to (close to a crud trap, heavily grinded)- **True but sufficient weld residual stress also exists at weld 5.**
- The AREVA report did not provide additional guidance related to the susceptibility of weld overlays that would lead one to conclude they should be looked at. **The lack of scope in a vendor report does not excuse licensee of responsibility for knowing or finding out extent of condition.**
- All industry experience pointed to the necessity of a heavy oxygenated environment or heavy stresses which weld #5 did not posses. **Statement below states that they had little industry operating experience so "all" is a misleading and likely inaccurate statement. Key is that the applicable industry OE does not identify a minimum oxygen concentration or amount of residual stress needed to initiate TGSCC than this was not a credible basis to exclude weld 5**.
- The understanding at the time was that high levels of oxygen provide the primary catalyst for TGSCC in nuclear plant primary coolant system, and they concluded this condition did not exist in weld #5 **Because the industry reports do not identify a minimum oxygen concentration or amount of residual stress needed to initiate TGSCC than this was not a credible basis to exclude weld 5.**
- Weld residual stresses alone if of sufficient magnitude, can drive a crack through-wall in a TGSCC environment (the licensee considered Weld #5 to have very low residual stresses due to it being an overlay and the location provided an easier environment for welding and the weld #5 had a smooth finish) **Welding residual stresses are very difficult to predict and this statement is an assumption not based upon credible analysis.**
- The 2001 root cause states that in comparison to the weld #3 cracking at Palisades, Fort Calhoun had essentially the same stress, very similar material, probably very similar chloride concentration, a lower temperature and a significantly higher dissolved oxygen concentration. This supports the licensee's position that oxygen played a heavy role in the cracking of weld #5 at Fort Calhoun a condition that was not believed to exist at Palisades. **Licensee made unsupported assumptions about minimum oxygen levels needed to support TGSCC.**

- Based on CRDM-21 taking 29 years to fail, the licensee concluded the current housings should last through the end of life of the plant, especially with the design changes made which would theoretically reduce the susceptibility of TGSCC for Weld #3 . Also at the point these decisions were being, made the licensee was intending to replace the Head as well as all the housings in 2006, an activity which was not performed due to the economic downturn. Bad assumption assuming that no other weld **These statements are not relevant because they do not evaluate/consider length of time to propagate TGSCC at other housing locations.**
- There was very little OE and industry guidance at this point in time and the information available discounted Weld #5 in a vented CRDM as being an area for concern. **Volume of OE is not as important as the applicability**

NRC's point

- The root cause points out that weld #5 is exposed to the same environment as weld #3
- The Fort Calhoun issue proved that the weld overlay could obtain a flaw that would propagate through wall.
- After changing the design they did not perform a susceptibility analysis based on the resulting stresses on all the welds as they had done prior to deciding to replace all the housings (still confirming this, requested documentation if it exists)
- The root cause points out that manufacturing played a heavy role in this issue yet in 2001 fabrication restrictions were not applied to weld #5 as they were to weld #4 and #3.
- In 2001 the root cause was determined to be TGSCC that occurred as a result of a susceptible material existing in an enabling environment under adverse stress conditions, and by not analyzing the stress conditions of Weld #5 in the post 2001 design the licensee could not make the determination that these characteristics did not apply to weld #5 (still confirming, requested supporting documentation to refute)
- Machining was categorized as an issue during the 2001 root cause and it was confirmed during the 2012 testing that machining was performed on CRDM 24 weld #5 that wasn't taken into consideration as far as what the residual stresses in the weld could be.
- By identifying weld #5 as less susceptible does not eliminative the susceptibility all together which means it should have been considered when developing an inspection plan.
- The licensee did not consider the effects of cold working on the stresses in weld #5.

- The 2001 root cause states that in comparison to the weld #3 cracking at Palisades, Fort Calhoun had essentially the same stress, very similar material, probably very similar chloride concentration, a lower temperature and a significantly higher dissolved oxygen concentration. This supports our point because it is mentioned on numeral occasions that the factors necessary for TGSCC propagation are environment, susceptible material and stress, which in accordance with this statement are very similar between Weld #3 and Weld #5.
- The licensee identified using Alloy 600 would provide resistance to IGSCC and TGSCC yet did not go with this option due to cost.
- In the organizational/programmatic weakness section of the 2012 root cause the licensee states (and tags as associated with RC1 and CC1) The 1991 Fort Calhoun OE was not adequately utilized to include inspections of the housing ID weld buildups.

My current position:

Based on my review of the 2001 and 2012 root causes (pending additional information requested) my position is that the licensee failed to perform evaluations and analyses of the stresses specifically for the weld overlay for the CRDM design installed in 2001 resulting in a failure to take corrective actions necessary to prevent recurrence.

Holmberg, Mel

From: Sanchez Santiago, Elba
Sent: Wednesday, March 13, 2013 9:11 AM
To: Giessner, John; Hills, David; Alley, David; Holmberg, Mel; Betancourt, Diana
Cc: Scarbeary, April; Taylor, Thomas; Shah, Swetha; Lennartz, Jay
Subject: CRDM Inspection status update

All,

There were a couple of issues we discussed during last Thursday's meeting. The following is a summary of each along with the **current status/recommendation**:

1. Crack growth rate URI
    a. After reviewing the licensee's root cause report we identified some discrepancies in the values they were using for crack growth rates. Specifically when describing the crack growth rate argument provided by the NRC they used an 11 month timeframe rather than the 24 month timeframe we had used in our calculations. This would bring the calculated crack growth rate for the postulated scenario much closer to the other calculated crack growth rates, making it a credible scenario and not an overly conservative assumption as described in the root cause report. **We brought this issue up to licensee and suggested they update their root cause report to accurately represent the crack growth rate associated with the described scenario.**
    b. In their root cause report the license described various theories and calculations which produced an array of possible crack growth rates. The licensee did not commit directly to any crack growth rate and the proposed inspection plan bounded the most conservative scenario by proposing inspections of a sample (10-12 housings) every refueling outage. Nonetheless it is mentioned in the inspection frequency section of the Inspection Plan that once a crack initiates it would take 4 to 5 cycles to propagate through wall. This to us represents a commitment to the least conservative crack growth rate and our concern is that in the future this could be a factor that to them justifies increasing the inspection frequency. **We communicated this to the licensee and communicated that closing the URI depended on what crack growth rate they were committing to and whether we agreed it was a conservative assumption. By including the least conservative crack growth rate in the inspection plan we don't feel comfortable closing the URI based on an Inspection Plan that is currently bounding but includes information that could change it to a less conservative plan. The licensee indicated they would either change that statement in the inspection plan and/or (Note: this was very loosely mentioned by a supervisor not senior management) commit on the docket to performing the inspections as currently stated in the inspection plan. Closing the URI will depend on the licensee's actions related to this issue.**
2. Prevent recurrence URI
    a. At our last call we discussed whether the conclusions reached by the licensee in 2001 on what actions to take concerning CRDM housing through wall leakage were reasonable based on the information they had at the time. After extensively reviewing the 2001 root cause report, 2012 root cause report as well as other supporting documents and having discussions with the licensee as well as our own technical experts **it is my current position and recommendation to issue a Criterion XVI violation for failure to prevent recurrence of CRDM leakage due to TGSCC. The following are some of the points that support my position:**
        i. The licensee identified in 2001 that residual stresses and machining played a key role in the initiation and propagation cracks through TGSCC and in the 2012 root cause report they mention fabrication stresses are developed principally from welding and metal working and the weld buildup on the inner surface was machined after welding.

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ii. They also stated in 2001 that the material and environment were conducive to TGSCC and that weld #3 and weld #5 were in essentially the same environment and composed of the same material.
iii. In the 2012 root cause they mention (and associate this statement with the current root cause and contributing cause) the 1991 Fort Calhoun OE was not adequately utilized to include inspections of the housing ID weld buildups.
iv. The actions the licensee took in 2001 to prevent recurrence focused on reducing the residual stresses and controlling the machining and surface finish conditions on welds 3 and 4. Because of their narrow focus the licensee failed to establish similar restrictions on the fabrication of weld 5, allowing residual stresses to remain in the weld and machining operations/cold work be applied to the weld without regards the resulting stresses in the weld, the environment it is exposed to and material susceptibility to TGSCC, thus failing to prevent recurrence of a through wall crack caused by TGSCC.

3. Inspection Plan scope concerns
   a. Based on the information obtained from both the current root cause report as well as the 2001 root cause report we are concerned the licensee is again applying a narrow focus to the issues and not including other susceptible welds (3 and 4) in their current inspection plan. The licensee stated that the fabrication method used (last pass heat sink welding) would ensure the ID of the weld is in compression rather than tension therefore preventing cracks from forming and propagating. We questioned whether there were tests, analyses or calculations performed that would demonstrate the net resulting stresses after taking into account hoop stresses induced during operation would be conservatively compressive. We also questioned whether there were weld repairs performed on welds 3 or 4 that would then eliminate the compressive stresses provided by the heat sink welding and result in tensile stresses. Basically we asked for the licensee to provide adequate justification for why welds 3 and 4 aren't a concern and shouldn't be included in the inspection plan. The licensee indicated the information is extensive and would take a large amount of time to compile and develop a justification. They also indicated they are contemplating performing visual examinations of welds 3 and 4 during the upcoming outage(very preliminary information). **Based on this I am recommending opening an Unresolved Item to capture this concern pending the licensee's compiling of information and developing a response. The licensee is aware we will need additional information on either their plans regarding welds 3&4 (inspections, calculations, analyses, etc.) in order to close out the URI and they indicated they would have this information prior to the upcoming outage.**
   b. During the call there were also concerns expressed regarding the methodology used for inspection and whether it would be adequate to detect flaws in the CRDM housing. The licensee indicated they will be performing demonstrations on the spare CRDM housings and that they are planning on accompanying the eddy current testing with visual examinations and ultrasonic testing as necessary. This information is also very preliminary. I requested the licensee keep us informed of their plans and demonstration schedule to ensure we are aware of what they consist of and have a chance to review and communicate our concerns to them prior to them implementing their plan this upcoming outage.

All of this information will be communicated to the site senior management today (Engineering Director). Therefore I will have additional information on the licensee's position regarding these issues during tomorrow morning's call. Let me know if you have any questions or concerns.

Thanks,
Elba

Holmberg, Mel

From: Sanchez Santiago, Elba
Sent: Thursday, March 14, 2013 4:35 PM
To: Holmberg, Mel; Alley, David; Giessner, John; Betancourt, Diana; Hills, David; Taylor, Thomas; Scarbeary, April; Shah, Swetha; Lennartz, Jay
Subject: Palisades Violation on not evaluating welds 3 & 4

With respect to the violation we debriefed this morning for the failure to follow the root cause procedure and provide adequate justification for not including welds 3 & 4 in the generic implications section the licensee is providing some pushback. The licensee stopped by and provided their arguments as to why there is no performance deficiency. To them, they understand they didn't document it as clearly as they could have in the root cause report, but they had considered welds 3 and 4 and discounted them due to the following reasons:

- The welds were Heat Sink Welded which is a process used to remove residual stresses and create compressive stresses on the ID of the weld, which would essentially prevent cracks from forming and propagating.
- The design of the weld already takes into account the operating hoop stresses and therefore an additional analysis that compares the compressive stresses created by the welding process. They mentioned Fort Calhoun did a finite element analyses that demonstrates the weld stresses are net compressive. We requested information to validate the Fort Calhoun analysis bounded the Palisades conditions.
- The surface finish of the weld was required by specification to be RMS 125 which would be a smooth finish eliminating any potential stress risers and reducing the potential from crack initiation sites.
- The licensee indicated they did not consider welds 3 and 4 susceptible to the factors identified in the root cause because they used a different process to fabricate the welds, and being ASME welds they are fabricated to a higher pedigree and standard than the overlay weld was.

Based on the licensee's arguments for what factors discount welds 3 and 4 being susceptible to TGSCC, we requested documentation that provided justification for the inclusion of those factors I the design fabrication and QA review process. Hence we requested:

- Design documentation that provides the basis for concluding the net stresses in the weld are compressive
- Design Specifications that demonstrate what was requested during fabrication
- Fabrication documents that would demonstrate if any rework was performed or if issues arose that would cause the weld to be left in a tensile state
- QA documentation to demonstrate a review of the housings was performed to verify the product met the required specifications

The licensee is aware of our concerns and they are currently digging up the necessary information resolve this issue. The current plan is to exit with a NCV of 10CFR app B Criterion V as we discussed today with the caveat that we are willing to review additional information and make any necessary changes to our position based on that review.

-Elba

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RELEASE IN ENTIRETY

Giessner, John

From: Sanchez Santiago, Elba
Sent: Friday, March 15, 2013 12:05 PM
To: Alley, David; Taylor, Thomas; Scarbeary, April
Subject: technical debrief notes
Attachments: Technical Debrief notes.docx

These are the points I am planning to cover during today's technical debrief. Please review and provide me your comments/ let me know if I should add anything other than those topics covered.

Thanks,
Elba

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This is the technical debrief for the follow-up review of the CRDM housing cracking from august 2012. All the items that will be discussed are pre-decisional and subject to management review.

The results of this inspection will be documented in the resident's quarterly report 2013002.

We performed the review in accordance with IP 71152

Technical Debrief:

1. Unresolved Item Crack growth rate
   a. Identify weakness in that in your inspection plan you indirectly commit to a crack growth rate by mentioning in your inspection frequency section that if a crack is identified it would take 4 to 5 cycles to propagate through wall and this is an assumption we view as non-conservative and we don't agree with it. I'd just like to caution you that if in the future you change your inspection plan to match that crack growth rate assumption we would have concerns with that.
   b. State the NRC's current position on crack growth rate 24 months
   c. *State the licensee's current inspection plan is bounding of the potential crack* growth rates
   d. Another potential weakness I would like to point out from your root cause report is the inconsistencies from one section to another. You point out that you haven't been able to pinpoint the exact cause yet you make the statement that CRD-24 is unique. This may be true but without verification, making that declarative statement isn't necessarily accurate and it has the potential to limit what you're looking at and for. I understand you need to draw the line somewhere but I would suggest you keep in mind that you don't have a smoking gun when contemplating what options you have going forward.

2. Unresolved Item Failure to prevent recurrence and technical specifications violation for operating with pressure boundary leakage.
   a. Based on our review of your 2001 root cause report, 2012 root cause report, various vendor documents and interviews with your staff we identified a performance deficiency
   b. PD Failure to recognize the susceptibility of weld # 5 to TGSCC and therefore not apply the level of scrutiny and corrective actions to this weld resulting in a failure to prevent recurrence of leakage in the CRDM housing due to TGSCC.
   c. More than minor because it adversely affects the initiating events cornerstone objective for not limit the likelihood of events that upset plant stability, specifically the cornerstone attribute of equipment performance.
   d. This is a violation of 10 CFR appendix B Criterion XVI Corrective Actions for failure to prevent recurrence of a significant condition adverse to quality.

e. Because we concluded there was a failure to prevent recurrence we will not be recommending discretion be granted for the technical specification violation for operating with pressure boundary leakage for greater than the LCO specified time, but rather than issuing to separate violations, we would combine the two into one violation with two examples.
f. We are not recommending a cross cutting aspect be applied to this performance deficiency because it occurred more than three years ago (11 years ago).

3. Violation Failure to follow the root cause procedure
   a. Through our review we raised concerns on the exclusion of welds 3&4 from your generic implications section in the root cause report and inspection plan. And though we understand you are compiling additional information to provide to us the position I am leaving with is as follows.
   b. PD Failure to adequately consider welds 3 and 4 in the generic implications portion of the root cause report and therefore provide justification for why no additional corrective actions associated with these welds are needed.
   c. What our concern is with this issue is that you may again be applying a narrow focus for what the potential of this cracking is. By essentially looking for like for like scenarios rather than considering what other portions of the components could be subject to the same or similar factors and whether the factors that discount them are valid based on research, analysis, review and or inspection.
   d. We consider this a violation of 10 CFR Appendix B Criterion V, for failure to follow procedures. Specifically the root cause procedure which requires you establish corrective actions for valid generic implications and if no corrective actions are proposed THEN document the rational.
   e. We categorized this issue as more than minor because if left uncorrected it has the potential to lead to a more significant safety concern Specifically if you don't have adequate justification for not including welds 3 and 4 in your generic implications section and have adequate justification for not taking corrective actions to address the potential generic concern it may result in another through wall leak.
   f. We are proposing a cross cutting aspect in the area of Human Performance, decision making for non-conservative assumptions. Specifically we believe you did not use non conservative assumptions when electing not to include welds 3 and 4 as a potential generic implication. But going forward we would be open to discussions on what you think the potential cross cutting aspect associated with this finding is.
   g. We understand what your position is on this issue and as mentioned before we will be reviewing the additional documentation you are providing us and would change the characterization of this issue as necessary depending on the results of our review. Do I say this or do I ask them their position on this issue? Also, do I ask them their position on the prevent recurrence issue?

h. The formal exit of this issue will be during the resident's quarterly exit and we will be in communication with you to inform you of any changes regarding this issue.

Hills, David

From: Holmberg, Mel
Sent: Monday, March 18, 2013 7:20 AM
To: Sanchez Santiago, Elba
Cc: Alley, David; Giessner, John; Betancourt, Diana; Hills, David; Taylor, Thomas; Scarbeary, April; Shah, Swetha; Lennartz, Jay
Subject: RE: Palisades Violation on not evaluating welds 3 & 4

Elba,

Sounds like you're on the right track to me. The performance deficiency "did not document the basis for excluding welds 3&4 within their root cause generic evaluation section" is still valid based on their feedback. I suspect (since they claim Fort Calhoun did an FE) that the licensee has not done an owner review and accepted under the licensee's design control process for FE analysis on this welds. Further, it is likely the FE work was completed by the vendor to help sell their fabrication process and it may not be directly applicable to the Palisades CRDM weld configuration nor is it likely they confirmed analysis by laboratory testing to measure residual weld stresses and validate the analysis and lastly it may not have considered the additional operating hoop stress that would offset weld compressive stress at the ID. Also, if they did any weld repairs it may undo the planned effectiveness of this process. So, I would not put too much hope on what they will be able to produce, but I agree it would be prudent to wait and see what they can deliver to determine if this violation is more than minor.

M

From: Sanchez Santiago, Elba
Sent: Thursday, March 14, 2013 4:35 PM
To: Holmberg, Mel; Alley, David; Giessner, John; Betancourt, Diana; Hills, David; Taylor, Thomas; Scarbeary, April; Shah, Swetha; Lennartz, Jay
Subject: Palisades Violation on not evaluating welds 3 & 4

With respect to the violation we debriefed this morning for the failure to follow the root cause procedure and provide adequate justification for not including welds 3 & 4 in the generic implications section the licensee is providing some pushback. The licensee stopped by and provided their arguments as to why there is no performance deficiency. To them, they understand they didn't document it as clearly as they could have in the root cause report, but they had considered welds 3 and 4 and discounted them due to the following reasons:

- The welds were Heat Sink Welded which is a process used to remove residual stresses and create compressive stresses on the ID of the weld, which would essentially prevent cracks from forming and propagating.
- The design of the weld already takes into account the operating hoop stresses and therefore an additional analysis that compares the compressive stresses created by the welding process. They mentioned Fort Calhoun did a finite element analyses that demonstrates the weld stresses are net compressive. We requested information to validate the Fort Calhoun analysis bounded the Palisades conditions.
- The surface finish of the weld was required by specification to be RMS 125 which would be a smooth finish eliminating any potential stress risers and reducing the potential from crack initiation sites.
- The licensee indicated they did not consider welds 3 and 4 susceptible to the factors identified in the root cause because they used a different process to fabricate the welds, and being ASME welds they are fabricated to a higher pedigree and standard than the overlay weld was.

Based on the licensee's arguments for what factors discount welds 3 and 4 being susceptible to TGSCC, we requested documentation that provided justification for the inclusion of those factors I the design fabrication and QA review process. Hence we requested:

- Design documentation that provides the basis for concluding the net stresses in the weld are compressive
- Design Specifications that demonstrate what was requested during fabrication

- Fabrication documents that would demonstrate if any rework was performed or if issues arose that would cause the weld to be left in a tensile state
- QA documentation to demonstrate a review of the housings was performed to verify the product met the required specifications

The licensee is aware of our concerns and they are currently digging up the necessary information resolve this issue. The current plan is to exit with a NCV of 10CFR app B Criterion V as we discussed today with the caveat that we are willing to review additional information and make any necessary changes to our position based on that review.

-Elba

Giessner, John RELEASE ENTIRETY

From: Sanchez Santiago, Elba
Sent: Monday, March 25, 2013 11:18 AM
To: Giessner, John; Betancourt, Diana
Subject: CRDM housing notes
Attachments: Technical Debrief notes.docx

Attached are my notes from my technical debrief at Palisades. Let me know if you have any questions.

-Elba

C20

1. Unresolved Item Crack growth rate
    a. The NRC notes that the conclusions section of the root cause report includes a conclusion from a B&W which indicates that the fracture surface contains beach marks but that it cannot be determined whether those beach marks relate to refueling outages or more frequent events, e.g., occurring over 24 months.
    b. Based on item 1 the NRC notes that the Root Cause appropriately addresses several potential corrosion rates.
    c. The NRC finds this your approach to addressing the uncertainty in corrosion rate acceptable because the inspection interval contained in your inspection plan bounds all the corrosion rates discussed, i.e., the current inspection interval is one refueling outage and all crack growth rates proposed require longer than one outage for a crack to grow from non detectible to through wall
    d. The NRC does note some weaknesses in your root cause / inspection plan related to crack growth rate These are:
        i. Crack growth rate discussion bases one crack growth rate on 6 outages occurring in 11 months. Given that not all of these outages resulted in pressure or heat up cool down cycles, a more appropriate time interval would be 24 months
        ii. Given the above cited error in the time interval for one of the crack growth rate calculations, the value cited in the root cause report is over estimated. When the correct time period is used, the calculated values is consistent with crack growth rates from other events. This crack growth rate should not be characterized as "overly conservative" as is currently the case
        iii. The crack growth rate based on refueling cycles appears to be under estimated. NRC inquires into the operation of the plant revealed that the CRDM housings were vented, allowing oxygen to enter, at least one time in addition to refueling outages. This was for an outage to replace seals on a reactor coolant pump. Additional opportunities for oxygen ingress may occur each time a seal housing is replaced. Based on these observations, the crack growth rate identified is understated by at least one refueling outage.
        iv. Although the inspection plan is designed to be bounding to the most rapid crack growth rate considered, the only mention of crack growth rate in the inspection plan is that a through wall crack requires 4 – 5 cycles to grow. This statement is inconsistent with the root cause evaluation and is considered non conservative. This statement could inadvertently result in a revision of the inspection interval to a non conservative value.

    e. Another potential weakness I would like to point out from your root cause report is the inconsistencies from one section to another. You point out that you haven't been able to pinpoint the exact cause yet you make the statement that CRD-24 is unique. This may be true but without verification, making that declarative statement isn't necessarily accurate and it has the potential to limit what you're looking at and for. I understand you need to draw the line somewhere but I would suggest you keep in mind that you don't have a smoking gun when contemplating what options you have going forward.

2. Unresolved Item Failure to prevent recurrence and technical specifications violation for operating with pressure boundary leakage.
    a. Based on our review of your 2001 root cause report, 2012 root cause report, various vendor documents and interviews with your staff we identified a performance deficiency
    b. PD Failure to recognize the susceptibility of weld # 5 to TGSCC and therefore not apply the level of scrutiny and corrective actions to this weld resulting in a failure to prevent recurrence of leakage in the CRDM housing due to TGSCC.
    c. More than minor because it adversely affects the initiating events cornerstone objective for not limit the likelihood of events that upset plant stability, specifically the cornerstone attribute of equipment performance.
    d. This is a violation of 10 CFR appendix B Criterion XVI Corrective Actions for failure to prevent recurrence of a significant condition adverse to quality. I can't call this a NCV as of yet, until it has been entered into your corrective action program.
    e. Because we concluded there was a failure to prevent recurrence we will not be recommending discretion be granted for the technical specification violation for operating with pressure boundary leakage for greater than the LCO specified time, but rather than issuing to separate violations, we would combine the two into one violation with two examples.
    f. The performance deficiency screened as green after screening under the initiating events cornerstone because we answered no to the question if after *reasonable assessment of degradation, could the finding result in exceeding the* RCS leak rate for a small LOCA and could the finding have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. Basically because of the nature of the cracking and your site procedures we believe it would be detected and corrected prior to reaching the small break LOCA limits.
    g. We are not recommending a cross cutting aspect be applied to this performance deficiency because it occurred more than three years ago (11 years ago).

3. Violation Failure to follow the root cause procedure
    a. Through our review we raised concerns on the exclusion of welds 3&4 from your generic implications section in the root cause report and inspection plan. And

though we understand you are compiling additional information to provide to us the position I am leaving with is as follows.

b. PD Failure to adequately consider welds 3 and 4 in the generic implications portion of the root cause report and therefore provide justification for why no additional corrective actions associated with these welds are needed.
c. What our concern is with this issue is that you may again be applying a narrow focus for what the potential of this cracking is. By essentially looking for like for like scenarios rather than considering what other portions of the components could be subject to the same or similar factors and whether the factors that discount them are valid based on research, analysis, review and or inspection.
d. We consider this a violation of 10 CFR Appendix B Criterion V, for failure to follow procedures. Specifically the root cause procedure which requires you establish corrective actions for valid generic implications and if no corrective actions are proposed THEN document the rational.
e. We categorized this issue as more than minor because if left uncorrected it has the potential to lead to a more significant safety concern Specifically if you don't have adequate justification for not including welds 3 and 4 in your generic implications section and have adequate justification for not taking corrective actions to address the potential generic concern it may result in another through wall leak.
f. The performance deficiency screened as green after screening under the initiating events cornerstone because we answered no to the question if after reasonable assessment of degradation, could the finding result in exceeding the RCS leak rate for a small LOCA and could the finding have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. Basically because of the nature of the cracking and your site procedures we believe it would be detected and corrected prior to reaching the small break LOCA limits.
g. We are proposing a cross cutting aspect in the area of Human Performance, decision making for non-conservative assumptions. Specifically we believe you did not use non conservative assumptions when electing not to include welds 3 and 4 as a potential generic implication. But going forward we would be open to discussions on what you think the potential cross cutting aspect associated with this finding is.
h. We understand what your position is on this issue and as mentioned before we will be reviewing the additional documentation you are providing us and would change the characterization of this issue as necessary depending on the results of our review. If the violation doesn't change and after the report is issued you want to contest this issue it will be delineated in the inspection report the process to use to do that.
i. The formal exit of this issue will be during the resident's quarterly exit and we will be in communication with you to inform you of any changes regarding this issue.

Holmberg, Mel

From:	Sanchez Santiago, Elba
Sent:	Thursday, March 28, 2013 3:54 PM
To:	Alley, David; Giessner, John; Hills, David; Holmberg, Mel
Subject:	FW: NRC Response
Attachments:	Weld 3 & 4 Justification.doc

All,

Attached is the Palisades response to the violation we debriefed on failure to comply with their root cause procedure and include welds 3 and 4 as part of the generic implications section. There will be a 10:30 am ET/9:30amCT call tomorrow to discuss our review of this document as well as the other documents provided. Feel free to contact me if you have any questions.

Thanks,

Elba M. Sanchez Santiago
Reactor Engineer
RIII/ DRS/ EB1
630-829-9715

From: Williams, Benjamin [mailto:bwill17@entergy.com]
Sent: Thursday, March 28, 2013 3:16 PM
To: Sanchez Santiago, Elba; Taylor, Thomas; Scarbeary, April; GUSTAFSON, OTTO W; DAVIS, TERRY A; Davis, Barry; Haumersen, Johannes; FOUTY, THOMAS H
Subject: NRC Response

Elba,

Attached is additional information concerning welds 3 and 4 for Palisades CRD Upper Housings.

Let me know if you have any questions.

Thanks,

Ben Williams
System Engineering
(269) 764-2196

C21

Palisades Nuclear Plant Upper Housing Pressure Boundary Weld: Justification That Welds Number 3 and 4 Were Considered During CRD-24 Leakage Investigation

Introduction

During a Nuclear Regulatory Commission (NRC) special inspection of the Palisades 2012 CRD-24 Upper Housing Leak, a question was raised as to why welds number 3 and 4 were excluded from the generic implications section of the root cause evaluation. The NRC was unable to locate any technical justification as to why corrective actions for the welds were not needed. A cross-cutting finding was proposed by the NRC for not including welds 3 and 4 in the generic implications section.

The purpose of this report is to identify areas of the Root Cause Evaluation that provide evidence that welds 3 and 4 were considered during the CRD-24 Root Cause Evaluation. Additional industry guidance is referenced to justify that acceptable methods were used to prevent reoccurrence of leakage from welds 3 and 4.

2001 Engineering Analysis: EA-EAR-2001-0426-01

In 2001, a leak from weld 3 was identified which was caused by Stress Corrosion Cracking and manufacturing irregularities. It was decided that a replacement of all the Upper Housings was necessary. A comprehensive engineering analysis performed by Palisades and Westinghouse (EA-EAR-2001-0426-01) was completed to update the Upper Housings to prevent Stress Corrosion Cracking from welds 3 and 4 and to determine the effects of the design changes.

As part of the CRD-24 Root Cause, the design changes that were made in 2001 to prevent reoccurrence of leakage from weld 3 and 4 were discussed. These improvements included:

- *"Application of heat sink welding. The heat sink welding is a proven technology in creating a compressive residual stress on the inside surface by water-cooling while performing the welding"* and *"Enhanced surface finishing by welding shrinkage and/or honing... the final finish was required to be RMS 125 or better". (CRD-24 Root Cause, pg 7).*

- Heat sink welding is a generally accepted method (1984 EPRI Research Project 1071-1) to reduce tensile stress on the interior of a weld. Making the inner diameter of a weld compressive would remove one of the elements of Transgranular Stress Corrosion Cracking and therefore would make the weld not susceptible to it.

- Engineering Analysis EA-EAR-2001-0426-01 was utilized for the Root Cause evaluation to determine that appropriate justification existed to eliminate the need for inspections of welds 3 and 4.

As part of the engineering analysis (EA-EAR-2001-0426-01) performed in 2001 to eliminate the risks of Stress Corrosion Cracking, a mockup of welds 3 and 4 were provided to Palisades for analysis by the manufacturer of the upper housings, Ionics.

Testing included:

- Visual examination, metallography, scanning electron microscopy, energy dispersive analysis, chemical analysis and hardness testing.

The metallurgical examinations (Consumers Energy, 2001, MAT Project: 0100642) performed under EA-EAR-2001-0426-01 concluded:

- *"No significant volumetric flaws were identified in the weld cross-section or adjacent base metal areas in either sample. No significant sensitization was observed". (Consumers Energy, pg 1).*

- Since no sensitization was detected in the welds, an element which was required for stress corrosion cracking, justification was provided that welds 3 and 4 did not need to be inspected.

Last Pass Heat Sink Welding Validation

An in-depth study of heat sink welding was completed by the Electric Power Research Institute (EPRI) in 1984 and concluded that:

- *"The results of this research project indicate that for pipe sizes on the order of 30.48 (12 inches) and less, LPHSW (Last Pass Heat Sink Welding) can effectively produce inside-diameter (ID) compressive residual stresses in the weld-heat-affected zone for all position welds". (EPRI 1984, pg iii).*

- This comprehensive study included destructive analysis and residual stress measurement both in the longitudinal and circumferential direction which gave Palisades additional confidence that the methods chosen to prevent leakage from reoccurring from welds 3 and 4 were valid.

2012 CRD-24 Root Cause Evaluation (CR-PLP-2012-5623)

From EN-LI-118, 5.[12].e:

- *"Perform an Extent of Cause evaluation by reviewing the individual Root and Contributing causes for generic implications".*

The root cause for the CRD-24 Upper Housing leak is:

- *"Stresses in the weld build up area due to manufacturing irregularities and misalignments between CRD-24 upper housing, support tube, seismic supports, and the associated reactor head penetration/CRDM nozzle". (CRD-24 Root Cause, pg 33).*

This uniqueness is based on the extent of condition inspections performed on 8 additional housings.

The contributing cause is:

- *"Transgranular Stress Corrosion Cracking (TGSCC) initiating within the internal weld buildup material of CRD-24". (CRD-24 Root Cause, pg 33).* Because of the uniqueness of the stresses in CRD-24, TGSCC was considered for the CRD Upper Housings extent of condition and extent of cause.

The extent of condition and extent of cause were based on the results of the Babcock and Wilcox destructive analysis of CRD-24 (PLP-RPT-12-000123). Included in the report was:

- *"Destructive examinations conducted on the nine (9) cracks identified during laboratory penetrant testing (PT) on the ID surface of the CRDM #24 housing." (CRD-24 Root Cause, pg 37).*

The pentetrant testing of the ID surface of the housing included the areas of welds 3 and 4. The upper housings were considered as a whole and therefore testing was conducted on the whole CRD-24 upper housing.

- **The dye penetrant testing of welds 3 and 4 did not show any indications of cracking even though the welds were exposed to the same conditions that promote Stress Corrosion Cracking as weld 5.**

Babcock and Wilcox concluded through destructive analysis that:

- *"Crack sizes ranged from 3" long (the thru wall crack at the "0" position) to 5/8" (lengths are approximate). All were noted to span or originate in the weld buildup area (see crack maps in the B&W report under RPT-PLP-12-000123). No circumferential cracks were identified as all identified cracks were axially located." (CRD-24 Root Cause, pg 37).*

- **Since cracking was identified only in the weld build up region in an upper housing that was known to have the conditions necessary for TGSCC to occur, it was acceptable to conclude the area around weld 5 was the only area necessary for additional inspections and welds 3 and 4 did not need inspections.**

Weld Repairs

During the CRD-24 Root Cause Analysis, all the weld repairs performed on the Upper Housings were identified and noted (CRD-24 Root Cause, pg 81-83). All of the repairs were completed in accordance with approved welding procedures as noted in the weld travelers. It was questioned whether or not the weld repairs defeated the advantages gained with last past heat sink welding.

- For all of the lower flange to pipe structure welds (weld 3), the welds were either cut out or excavated from the OD and then replaced in accordance with the approved weld procedure which would preserve the advantages of LPHSW.

Some of the upper flange to pipe structure welds (weld 4) required repair and were excavated from the ID then repaired which would increase the probability of defeating the advantages of LPHSW.

- The CRDs that were repaired in this manner include: CRD-7, 12, 17, 21, 26, 29, 30, 32, and 42.

- At the time of the root cause these repairs were not considered in the generic implications but further investigation revealed that the weld repairs are not at an increased risk for Stress Corrosion Cracking.

- The water around weld 4 is at about 250 deg F and the water at weld 5 is at about 530 deg F. This makes weld 4 less susceptible to SCC than weld 5.

In the 2001 Engineering Analysis, the decision whether to use 316 or 347 SS was discussed. It was noted that:

- Using a low carbon 316 stainless steel would help to prevent stress corrosion cracking.

EPRI also published a study in 1981 that predicted the critical cooling rate that would cause sensitization during welding. The report states:

- *"....when the carbon content is reduced to less than 0.03 wt% the critical cooling rate is predicted to be less than 0.5 deg C/s. For a 0.35 in plate such a cooling rate can only be exceeded by heat inputs as large as 3937 j/mm (100,000 J/in.) A realistic heat input of 984.3 J/mm (25,000 J/in) yields a cooling rate which is 40 times larger than 0.5 deg C/s, and thus no sensitization should be (or is) noted". (EPRI 1981, pg 2-18).*

The chemical analysis of the 316SS provided in the welding travelers from 2001 for the CRD upper housings resulted in a carbon content of about 0.016 %wt.

- This is much less than 0.03 %wt which would allow much higher heat inputs to be used before sensitization occurred. **Since the welding procedure only allows a maximum heat input of 45 KJ/in, there is not an opportunity for the metal to be sensitized.**

- Therefore, even with a weld repair at weld 4, there is no sensitization of the weld and a factor of stress corrosion cracking is removed.

Conclusion

Based on the following information, it can be concluded that Palisades considered welds 3 and 4 during the 2012 CRD-24 Root Cause Evaluation;

2001 Engineering Analysis of the redesigned Upper Housing:

- 2001 design changes (EA-EAR-2001-0426-01), including the improvements made to welds 3 and 4, were discussed in the Root Cause Evaluation.
- Improvements discussed included the compressive forces provided by Last Pass Heat Sink Welding and a RMS 125 surface finish which provided welds that are highly resistant to TGSCC.
- A comprehensive metallurgical analysis was also performed as a part of the engineering analysis (EA-EAR-2001-0426-01) to ensure that welds 3 and 4 would perform as required (Consumers Energy, 2001, MAT Project: 0100642).

Last Pass Heat Sink Welding:

- A 1984 EPRI study determined that Last Pass Heat Sink Welding was a valid and reliable way to produce compressive stresses on the interior of the weld. This weld process was followed in the manufacturing of welds 3 and 4.
- Testing included residual stress measurements to ensure that the welds were compressive.

2012 CRDM Root Cause Evaluation

- The generic implications section was based on the root and contributing causes for CRD-24.
- **Welds 3 and 4 were designed to prevent SCC (EA-EAR-2001-0426-01) using an industry accepted method (EPRI, 1987, Research Project T109-2).**
- Weld 5 was NOT designed to limit the sensitivity to TGSCC.
- The possibility that TGSCC could affect welds 3 and 4 was considered so the entire ID of CRD-24 was penetrant tested by Babcock and Wilcox. No indication of cracking was found.
- All cracking was found at weld 5, within the weld buildup area.
- No cracking was found at welds 3 and 4 in 2012 on CRD-24.

- **An environment that was conducive to TGSCC was known to exist in CRD-24. Welds 3, 4 and 5 were exposed to this environment with cracking only being found in weld 5.**
- Cracking at weld 4 was not found in the 2001 destructive analysis of the Upper Housings.

- References to the Engineer Analysis and Babcock and Wilcox were noted in the Root Cause Analysis.

- Remote visual examinations are being developed to inspect welds 3 and 4.

Weld Repairs

- As discussed in the 2001 Engineering Analysis, low carbon 316 SS was used for the Upper Housings

- Type 316SS with less than a 0.03 wt% carbon needs a very high heat input rate during welding for the material to become sensitized

- Palisades Upper Housings has 0.016 wt% which requires an extremely large amount of heat for the material to sensitize

- Weld heat inputs were limited to 45 KJ/in.

- ID weld repairs at weld 4 are not sensitized

Additional Information

As part of Palisades review of options for inspecting the Upper Housings during the fall 2013 refueling outage, the ability to perform remote visual examinations was requested in January of 2013 from Westinghouse in addition to eddy current testing. The visual examination will be used to examine welds 3, 4 and 5 during the inspections of the 12 CRDM housings selected for the upcoming refueling outage and will allow cracking to be identified in welds 3 and 4. Of the 12 Upper Housings included in the inspection plan for 1R23, 3 have had ID weld repairs at weld 4 which will validate the justification provided in this document.

Babcock and Wilcox took extensive photos of the CRD-24 Upper Housing during destructive analysis including the cracks in weld 5 before penetrant testing. The cracks at weld 5 in the CRD-24 Upper Housing were able to be distinguished in as-found photos taken with a digital camera. Therefore, Palisades has high confidence that if cracking in welds 3 and 4 has occurred then they can be easily found through visual exams.

References

2001 Engineering Analysis – EA-EAR-2001-0426-01

Babcock and Wilcox – Examinations of Cracks in CRDM #24 Housing (PLP-RPT-2012-00123)

Basic Studies on the Variabilities of Fabrication-Related Sensitization Phenomena in Stainless Steels – EPRI, 1981, Research Project 1071-1

Last Pass Heat Sink Welding Process Development – EPRI, 1987, Research Project T109-2

Palisades Metallurgical Examination of CRD Weld Samples – Consumers Energy, 2001, MAT Project: 0100642

Root Cause Evaluation Report: CRD-24 Upper Housing Leak – CR-PLP-2012-05623

Sanchez Santiago, Elba

From: Giessner, John
Sent: Friday, March 29, 2013 4:05 AM
To: Sanchez Santiago, Elba; Alley, David; Holmberg, Mel; Hills, David; Taylor, Thomas; Scarbeary, April; Betancourt, Diana
Subject: Re: NRC Response

To consider:
Did they consider max tensile stresses at pressure and thermally?

Before we call it a weakness we should ask our confidence in weld 3/4. Recall the code requires ndt only once a isi cycle(and then only a % of periphry). They haven't been done -since install in 2001, and only crd-24 had an NDT on weld 3 and only a DPT last year. And none will be done this outage.Right? Is our confidence that good. If it is, it is a weakness;if we still need action, it's a ncv.
(Sent from Blackberry)

From: Sanchez Santiago, Elba
To: Alley, David; Giessner, John; Holmberg, Mel; Hills, David; Taylor, Thomas; Scarbeary, April; Betancourt, Diana
Sent: Thu Mar 28 21:21:05 2013
Subject: RE: NRC Response

Attached is a portion of the metallurgical report the licensee provided me (Dave Alley, this is the report I said I'd send you but when I tried scanning in the second portion, it kept failing. I will try again tomorrow morning) I also attached some specific information provided with regards to their welding process.

In regards to the white paper, the licensee mentions that the process used was Last Pass Heat Sink Welding (LPHSW) and they provide EPRI information related to this method. In accordance with the attached document (weld process) the process was indeed heat sink welding, but not LPHSW. I don't know how many differences there are between heat sink welding and LPHSW and whether it matters which one they used but I do know in an ASM document they are described separately and though the purpose of both is to reduce the stresses in the weld, LPHSW is specifically called out as leaving the weld in compressive stress where heat sink welding is described as leaving the weld either in very low tensile stress or compressive stress (my point being that at least in the ASM document the description for heat sink welding wasn't as definitive as for LPHSW)

Also of note is that the traveler used for performing the repairs notes that the repair is only performed using water backing (essential for LPHSW) when the repair extends through the pressure boundary. If the weld repair is performed from the OD (weld #3) and it is not through wall, can the grinding process used to excavate the weld induce residual stresses in the ID? Is heat sink welding necessary to ensure the weld is left in a compressive state after the repair?

In regards to Dave's comments below, if the justification addresses our concerns I would lean towards calling the issue a weakness identified in their root cause analysis. My rationale for this would be that if the information provided is enough, then the issue would be more a thorough documentation issue rather than a concern that the licensee did not adequately address the potential generic implications.

C22

From: Hills, David
Sent: Thursday, March 28, 2013 4:24 PM
To: Holmberg, Mel; Sanchez Santiago, Elba
Cc: Alley, David; Giessner, John
Subject: RE: NRC Response

Of course, I don't think the issue was entirely what type of justification can they eventually come up with, it was the analysis/justification they had in place at the time we did the inspection. However, if they eventually develop adequate justification in response to our concerns, then the question becomes how to we distinguish between this being a finding/violation versus just one of several weaknesses already identified in their root cause analysis.

From: Holmberg, Mel
Sent: Thursday, March 28, 2013 4:11 PM
To: Sanchez Santiago, Elba
Cc: Alley, David; Giessner, John; Hills, David
Subject: RE: NRC Response

Elba, I did a quick read. Seems they have a good story for weld No. 3. I am still uncertain about their technical basis to exclude weld No. 4 welds with repairs.
I noticed that they <u>did not</u> compare the heat input used for weld No. 5 and compare this with heat input for weld repairs to weld No. 4 (45 kj/in). If heat input allowed for weld 5 was substantially higher than allowed for the weld repairs on weld 5 than their story makes sense.

M

From: Sanchez Santiago, Elba
Sent: Thursday, March 28, 2013 3:54 PM
To: Alley, David; Giessner, John; Hills, David; Holmberg, Mel
Subject: FW: NRC Response

All,

Attached is the Palisades response to the violation we debriefed on failure to comply with their root cause procedure and include welds 3 and 4 as part of the generic implications section. There will be a 10:30 am ET/9:30amCT call tomorrow to discuss our review of this document as well as the other documents provided. Feel free to contact me if you have any questions.

Thanks,

Elba M. Sanchez Santiago
Reactor Engineer
RIII/ DRS/ EB1
630-829-9715

From: Williams, Benjamin [mailto:bwill17@entergy.com]
Sent: Thursday, March 28, 2013 3:16 PM
To: Sanchez Santiago, Elba; Taylor, Thomas; Scarbeary, April; GUSTAFSON, OTTO W; DAVIS, TERRY A; Davis, Barry; Haumersen, Johannes; FOUTY, THOMAS H
Subject: NRC Response

Elba,

Attached is additional information concerning welds 3 and 4 for Palisades CRD Upper Housings.

Let me know if you have any questions.

Thanks,

Ben Williams
System Engineering
(269) 764-2196

Sanchez Santiago, Elba

From: Holmberg, Mel
Sent: Monday, April 15, 2013 4:08 PM
To: Sanchez Santiago, Elba
Subject: RE: Palisades inspection report input
Attachments: Palisades Input to DRP Report 2013 002 URI EMS -msh comments.docx

Elba,

Only got thru your first finding today. Here is what I have so far, and I will try to get thru other finding by tomorrow.

M

From: Sanchez Santiago, Elba
Sent: Monday, April 15, 2013 12:48 PM
To: Holmberg, Mel
Subject: Palisades inspection report input

Mel,

I attached the draft palisades inspection report input in case you have a chance to review and provide me your input.

Thanks,

Elba M. Sanchez Santiago
Reactor Engineer
RIII/ DRS/ EB1
630-829-9715

C/13

UNITED STATES
NUCLEAR REGULATORY COMMISSION
LISLE, IL 60532-4352

April XX, 2012

MEMORANDUM TO: Thomas Taylor
Senior Resident Inspector
Palisades Nuclear Plant

FROM: David Hills, Chief
Engineering Branch 3
Division of Reactor Safety

SUBJECT: PALISADES NUCLEAR PLANT
DRS INPUT TO INTEGRATED REPORT 05000255/2013002

Enclosed is the report input for the Palisades Nuclear Plant, Inspection Report 05000255/2013002. This report input documents completion of our review of Unresolved Items 05000255/2012012-01, "TS for PCS Pressure Boundary Leakage," 05000255/2012012-02, "Potential Inadequate Degradation Evaluation of CRDM Housings," and 05000255/2012012-03, "Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality." This report also completes the one sample of the Problem Identification and Resolution, Selected Issue Follow-up in accordance with IP 71152. I have reviewed this input to confirm compliance with Inspection Manual Chapter (IMC) 0612 and IMC 305. This input is ready for inclusion into the integrated report and dissemination to the public.

Please input the following post Inspection Data into RPS:

Inspection Procedure	**Procedure Status** – see below: Incomplete, Complete, Complete by reference, Complete-full sample not available, Complete – opportunity to apply procedure not available, Not Applicable.	**Sample Size** – As documented in Scope Section If less than full sample size documented in the report input, the inspector must provide a justification below to enter into RPS and support the procedure status selected
71152	Complete	1

Inspection Report Item and Type (AV, FIN, NCV, URI or VIO)	**Cornerstone** (IE, MS, BI, EP, OR, PR, MISC)	**Cross Cutting Aspect** (H.n(i), P.n(i), S.n(i))	**Responsible Person/Owner**	**Procedure or TI** (71111.07T)	**RPS Branch Code (e.g. closeout responsibility)** EB1 3820 EB2 3870 EB3 3840 PST (RP) 3860 PSB (Safeguards) 3850 OB 3810
NCV-XXX	IE	n/a	E. Sanchez Santiago	71152	3820
NCV-XXX	IE	H.1(b)	E. Sanchez Santiago	71152	3820

Enclosure: Input to Inspection Report 05000255/2013002

cc w/encl: J. Giessner, Chief
C. Hernandez, Site Admin Assistant

CONTACT: E. Sanchez Santiago, DRS
(630) 829-9715

DOCUMENT NAME: G:\DRSIII\DRS\Work in Progress\-Palisades Input to DRP Report 2013 002 URI EMS.docx
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OFFICE	RIII		NRR		RIII		NRR	
NAME	ESanchezSantiago		DAlley		DHills		TLupold	
DATE	4/ /13							

OFFICIAL RECORD COPY

Cover Letter

_X_ Green findings involving a violation were identified. Include the following:

Based on the results of this inspection, two NRC-identified findings of very low safety significance (Green) were identified. These findings were determined to involve a violation of NRC requirements. However, because of the very low safety significance and because the issues were entered into your corrective action program, the NRC is treating the issue as Non-Cited Violation, in accordance with Section 2.3.2 of the NRC Enforcement Policy.

TITLE PAGE

Inspectors: D. Alley, Senior Materials Engineer
E. Sanchez Santiago, Reactor Inspector

SUMMARY OF FINDINGS

A. NRC-Identified and Self-Revealed Findings

Cornerstones: Initiating Events

- Green. The inspectors identified a Non-Cited Violation (NCV) of 10 CFR Part 50 Appendix B Criterion XVI, and Technical Specification (TS) 3.4.14 Primary Coolant System (PCS) Operational Leakage, for failure to prevent recurrence of a significant condition adverse to quality resulting a non-compliance with the TS. Specifically, the licensee failed to take adequate corrective actions in response to a pressure boundary leak from CRDM housing in 2001 which resulted in a pressure boundary leak from a similar CRDM housing in August 2012. The licensee operated with this pressure boundary leak for greater than the TS allowed time. The licensee entered this issue into their corrective action program as CR-PLP-2013-01134.

  The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. Specifically the licensee did not limit the likelihood of events that upset plant stability by not taking adequate corrective actions to prevent recurrence of leakage in CRDM housings which represents a pressure boundary leakage and a condition prohibited by the technical specifications. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the because the failure of a CRDM housing is a Primary System LOCA initiator contributor. In accordance with Table 3 "SDP Appendix Router" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors answered "no" to all the questions in Sections A through E, and were directed to IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power." The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The

Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors determined that finding was not indicative of current performance and therefore a cross-cutting aspect was not applied. (Section 4OA2.b(1))

- Green. The inspectors identified a Green Non-Cited Violation (NCV) of 10 CFR Part 50 Appendix B Criterion V, for failure to follow the root cause procedure. Specifically, the licensee failed to adequately evaluate the generic implications of the cause of the cracking identified in CRDM #24. The licensee entered this issue into their corrective action program as CR-PLP-2013-01500.

  The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. The inspectors also answered "yes" to the More-than-Minor screening questions "if left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, the licensee did not limit the likelihood of events that upset plant stability by not adequately evaluating the potential generic implications associated with welds 3 and 4, which could potentially result in another through wall leak. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the because the failure of a CRDM housing is a Primary System LOCA initiator contributor. In accordance with Table 3 "SDP Appendix Router" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors answered "no" to all the questions in Sections A through E, and were directed to IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power." The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors determined that the primary cause of the failure to adequately consider welds 3 and 4 on the generic implications section of the root cause report related to the cross-cutting component of Human Performance, Decision Making, because licensee staff did not use conservative assumptions in decision making. Specifically, the licensee did not use conservative assumptions when excluding welds 3 and 4 as being susceptible to TGSCC and therefore include them in the generic implications section of the root cause report. (Item H.1(b)). (Section 4OA2.b(2))

B. Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAILS

4. REACTOR SAFETY

4OA2 Identification and Resolution of Problems (71152)

.3 Selected Issue Follow-up Inspection: Through Wall Leakage of Control Rod Drive Mechanism (CRDM) Housing #24

a. Inspection Scope

On August 12, 2012 the licensee shutdown to investigate an increase in unidentified leakage. The source of the leakage was determined to be a crack in control rod drive mechanism housing (CRDM) No. 24. Shortly after the discovery of the leak in CRDM housing No. 24, the NRC dispatched a special inspection team (SIT) to review the CRDM No. 24 leakage event. The SIT identified an unresolved item (URI) related to the potential failure to prevent recurrence of a significant condition adverse to quality (SCAQ) which was considered an unresolved item, because the licensee's root cause investigation was ongoing at that time. The licensee subsequently removed the failed housing from service for further testing and completed an evaluation to determine the cause of the cracking (reference root cause report xxx).

From xxx to yyy, the inspectors completed one inspection sample regarding problem identification and resolution based upon review of the licensee's root cause report xxx and associated corrective action records related to this issue:

- CR

The inspectors reviewed the licensee's actions in accordance with performance attributes identified in IP 71152. Specifically, the inspectors reviewed licensee corrective action records to determine if: (1) the problems were accurately identified; (2) operability and reportability were adequately ascertained; (3) extent of condition and generic implications were appropriately addressed; (4) classification and prioritization of problem was commensurate with safety significance; (5) root and contributing causes were identified; (6) corrective actions were appropriately focused to correct problem; and (7) timely corrective actions were completed or proposed commensurate with the safety significance of the issues.

~~As a follow-up to this issue the NRC performed an inspection of the actions taken by the licensee in response the CRDM housing through wall crack. The inspection consisted of a review of the root cause report as well as supporting documentation provided by vendors, such as calculations and technical evaluations. The inspectors also reviewed available operating experience related to this issue.~~

b. Findings

.1 Failure to Prevent Recurrence of a Significant Issue Adverse to Quality

Introduction: The inspectors identified a Green Non-Cited Violation (NCV) of 10 CFR Part 50 Appendix B Criterion XVI, and Technical Specification (TS) 3.4.14 Primary Coolant System (PCS) Operational Leakage, for failure to prevent recurrence of a significant condition adverse to quality resulting a non-compliance with the TS. Specifically, the licensee failed to include the internal CRDM weld build-up area within the scope of corrective actions taken for a 2001 CRDM housing leakage event (a significant condition adverse to quality) and consequently leakage recurred at the CRDM housing No. 24 in 2012.

~~take adequate corrective actions in response to a pressure boundary leak from CRDM housing in 2001 which resulted in a pressure boundary leak from a similar CRDM housing in August 2012. The licensee operated with this pressure boundary leak for greater than the TS allowed time.~~

Description: On August 12, 2012 Palisades Nuclear Power Station shutdown to investigate an increase in unidentified leakage. During a walk-down performed post shutdown the licensee discovered the source of the leakage to be a pressure boundary leak from ~~Control Rod Drive Mechanism~~ (CRDM) Housing No.#24. After further testing, the licensee ~~it was~~ determined the leak ~~was associated with~~ occurred because of through-wall flaw adjacent to a weld build up on the interior of the housing. The purpose of this weld build-up is to maintain the ~~control drive mechanism~~ CRDM properly aligned. The root cause team (RCT) was composed of xxx site and vendor staff that conducted the root cause investigation in accordance with site procedures procedures xx and yy and on xx, issued a root cause analysis report zzz. In this report, the licensee's RCT ~~The licensee performed a root cause analysis to~~ determined that the probable cause of the cracking was "Stresses in the weld build up area due to manufacturing irregularities and misalignments between CRD-24 upper housing, support tube, and the associated reactor head penetration/CRDM nozzle. Based on lack of cracking found in the other 8 upper housings tested, the failed CRD-24 upper housing contains an as-yet unidentified additional stress". The ~~report~~ RCT also identified the following contributing cause: "Transgranular Stress Corrosion Cracking (TGSCC) initiating within the internal weld build-up material of CRD-24. The through wall crack initiated in the weld material and then propagated through the base metal until a leak developed in the OD witness band region at the base of the inner diameter (ID) weld build up. ~~The investigation consisted of performing~~ This conclusion was based upon ~~testing on the failed CRDM housing which included~~ destructive and non destructive examinations (NDE) ~~and as well as destructive analyses~~ completed on a section of the failed housing which included the through-wall flaw. The ~~licensee's~~ RCT ~~investigation~~ also ~~obtained vendor~~ relied upon vendor technical reports assessing ~~on~~ the results of the NDE ~~examinations~~ as well as vendor calculations related to the stresses in the CRDM housings.

~~In order~~ To determine the extent of condition, the licensee performed ultrasonic (UT) examinations ~~testing~~ of the weld build up area on 8 additional CRDM housings. ~~The results of the root cause analysis were documented in a root cause report which was reviewed by the NRC inspectors during the follow-up inspection. The root cause report defined the probable root cause as:~~

~~An event similar to this occurred~~ in 2001 ~~when~~ the licensee discovered a steam leak in the housing of CRDM-21 caused by a through-wall TGSCC at CRDM housing weld No. 3 which was located just below the weld build-up region. ~~which was also classified as pressure boundary leakage. In this case the crack was associated with a butt weld~~

~~located just below the aforementioned weld buildup.~~ This issue was categorized as a significant issue adverse to quality by the licensee (reference CR xx) and the licensee's root cause evaluation was documented in RCR/CR-xxx ~~performed~~. The ~~root cause evaluation~~ licensee concluded that the cracks in CRDM-21 were caused by TGSCC which occurred in areas of heavy grinding or machining tool marks. Specifically, this leak ~~in the housing~~ was the result of an inner diameter initiated, axially oriented, transgranular crack in the austenitic stainless steel housing material. The licensee's extent of condition investigation identified TGSCC at most of the inservice housings near the weld No. 3 location. The licensee's corrective actions ~~taken by the licensee~~ included replacing all 45 CRDM housings with modified housings. ~~when subsequent testing indicated additional cracks in the same location in other CRDM housings~~. The modifications that the licensee made to the replacement housings to prevent recurrence included: ~~controlling the fabrication process for the pressure retaining welds in the CRDM housing to ensure~~ prohibiting grinding ~~was not performed and the~~ at the ID surface during fabrication so that tensile residual stresses on the internal surface of weld were reduced; ~~The licensee also modified the physical~~ changing the ~~design~~ location of weld No. 3?? to reduce the design stresses and the accumulation of contaminants in proximity to the weld; ~~where the cracking had occurred. The licensee also~~ and changing the material of the housing from type 347 to type 316 stainless steel ~~component., though the replacement material was essentially equal to the previous material when comparing susceptibility to TGSCC. the actions specified by the licensee~~ to prevent recurrence of ~~cracking in the CRDM housings.~~

To evaluate the effectiveness of these preventative actions from the 2001 CRDM leakage event, the inspectors reviewed the licensee actions to determine if they had been sufficient to eliminate one of the 3 necessary factors to cause TGSCC on the CRDM housings: (1) a susceptible material, (2) a corrosive environment and (3) tensile stress." The inspectors identified that the licensee had failed to eliminate one or more of the necessary factors at the weld build-up area to preclude TGSCC in the replacement housing. Specifically:

- The licensee's 2001 root cause report documented that the weld build-up region is exposed to essentially the same environment as the weld that experienced the cracking (corrosive environment remained unchanged).
- No analysis was completed on the stress conditions for the weld build-up region prior to approving the modified replacement housing design (left residual tensile weld stresses on ID of CRDM surface).
- Fabrication restrictions to prohibit grinding were not applied to the weld build-up region (grinding promotes residual tensile stress state on ID of CRDM surface)
- Cold work (??? What specific type of cold work??)was applied to weld buildup areas during fabrication (induced residual tensile stresses)
- Material was changed from type 347 to type 316 stainless steel (both materials are essentially equally susceptible to TGSCC).

Based upon the recurrence of through-wall leakage in the CRDM housings that occurred at the weld buildup region of the CRDM housings, the inspectors concluded that the licensee actions had not been sufficient to preclude recurrence of TGSCC. Further, the actions to preclude recurrence were within the licensee's ability to foresee and prevent. Specifically, in 1991, the Fort Calhoun plant had experienced through-wall leakage due to TGSCC at the weld build-up region of their CRDM housings (same housing design) and this operational experience had been reviewed by the licensee and dismissed. Specifically, the licensee discounted the weld build-up region failure at Fort Calhoun because it occurred in the spare housings which they assumed had a more aggressive environment than the Palisades operating housings. In the licensee's 2012 RCR the RCT also concluded that due to organizational/ programmatic weakness ~~section of the 2012 root cause evaluation the licensee states~~ the 1991 Fort Calhoun operating experience was not adequately utilized to include inspection of the housing ID weld build-up regions.

~~Through their review of the 2001 root cause report, the inspectors noted the corrective actions were focused on the pressure retaining welds contained in the CRDM housing and due to various reasons the weld build-up region was excluded from the analyses and technical assessments performed in response to the through wall leak. The licensee also considered operating experience which included an incident at Fort Calhoun where a through-wall crack had developed in the weld build-up region of their CRDM housings. Fort Calhoun is the only additional plant to have the same CRDM housing design as Palisades. When making the comparison the licensee discounted the weld build up region because it did not meet the exact characteristics of the Fort Calhoun incident.~~

~~The inspectors also had various discussions with the licensee to address questions and concerns related to this issue. The activities performed by the inspectors also included internal discussions with regional inspectors and supervisors as well as technical experts from headquarters. Through their review of the information available and the internal and external discussions,~~

During the special inspection the inspectors also identified an unresolved item for the Technical Specification pressure boundary leak. The licensee determined the leakage commenced on July 14, 2012. The licensee operated in this condition for greater than 6 hours, which is the required shutdown time when pressure boundary leakage exists in the plant. Based on the information provided above, unresolved items 05000255/2012012-01 "TS for PCS Pressure Boundary Leakage" and 05000255/2012012-03 "Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality," are being closed to the following finding and associated violation.

Analysis: The inspectors determined that the licensee's failure to prevent recurrence of TGSCC of the CRDM housings (a significance condition adverse to quality) that resulted in a TS non-compliance was a performance deficiency that warranted a significance evaluation. ~~Specifically, the licensee failed to recognize the susceptibility of the weld build up region to TGSCC and therefore did not apply the level of scrutiny and corrective actions to this weld resulting in a failure to prevent recurrence of leakage in the CRDM housing due to TGSCC.~~ The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. Specifically the licensee did not limit the likelihood of events that upset plant stability by not taking adequate corrective actions to prevent recurrence of leakage in CRDM housings which represents a pressure boundary leakage and a condition prohibited by the technical specifications. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

~~In accordance with Table 3 "SDP Appendix Router" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors answered "no" to all the questions in Sections A through E, and were directed to IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power."~~ The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a large component rupture.

The inspectors determined that finding is not indicative of current performance and therefore a cross cutting aspect is not applied. (why ??? explain)

Enforcement: The inspectors identified a NCV of 10 CFR Appendix B Criterion XVI "Corrective Actions", and Technical Specification 3.4.13 "Primary Coolant System Operational Leakage", having a very low safety significance (Green), for failure to prevent the recurrence of leakage in CRDM housings due to TGSCC resulting in the operation of the reactor with pressure boundary leakage for greater than the TS allowed time.

Title 10 CFR Appendix B Criterion XVI requires, in part, that "In the case of significant conditions adverse to quality, the cause of the condition is determined and corrective action taken to preclude repetition."

Technical Specification 3.4.13 PCS Operation Leakage states, in part, "PCS operational Leakage shall be limited to no pressure boundary leakage." Condition B requires the licensee be in Hot Standby in 6 hours and Cold Shutdown in 36 hours when pressure boundary leakage exists.

Contrary to the above, from June 2001 through xxx 2012, the licensee failed to take adequate corrective actions to prevent recurrence of pressure boundary leakage in CRDM housings due to TGSCC. The leakage of CRDM housing # 21 identified in 2001 was categorized as a significant condition adverse to quality in accordance with the licensee's corrective action program. The licensee performed a root cause evaluation that determined the a contributing (need to check with EICS to see if it matters that this was not the root cause???) cause to be TGSCC. The corrective actions to prevent recurrence included changing the design to reduce stresses in the failed weld, control the surface finish of the pressure retaining welds to reduce potential crack initiation points and the welding process was also changed to reduce the stresses in the weld. These corrective actions were narrowly focused on the pressure retaining welds of the CRDM housings. As a result on August 12, 2012 a leak was identified from CRDM housing #24. The cause of this leak was also determined to be TGSCC. The source of

the leakage was specifically the weld build up region, which was ~~inadequately~~ inappropriately excluded from the scope of corrective actions taken in 2001 to prevent recurrence. The pressure boundary leakage was identified due to an increase in unidentified leakage noted on July 14, 2012. The plant did not enter Hot Standby until August 12, 2012 indicating the licensee operated with pressure boundary leakage for greater than the TS allowed time of 6 hours. (Need to identify actions proposed to restore compliance with criterion XVI???)
Because of the very low safety significance and because the licensee entered this issue into their corrective action program (CR-PLP-20136-01134), it is being treated as a NCV consistent with Section 2.3.2 of the Enforcement Policy (NCV 05000255/2013002-xx).

.2 Failure to Adequately Address the Generic Implications of the Cracking identified in CRDM 24

Introduction: The inspectors identified a Green Non-Cited Violation (NCV) of 10 CFR Part 50 Appendix B Criterion V, for failure to follow the root cause procedure. Specifically, the licensee failed to adequately evaluate the generic implications of the cause of the cracking identified in CRDM #24.

Description: While reviewing the root cause report related the cracking identified in CRDM #24 the inspectors identified a concern related to the generic implications of the cracking. The licensee's actions going forward were solely focused on the weld build up region, which is where the current failure occurred. The inspectors questioned why the other welds (weld #3 below the weld build up region and weld #4 above the weld build up region) contained in the CRDM housing were not included in the proposed corrective actions going forward. The licensee provided additional information to justify excluding these welds from the scope of the corrective actions. The NRC inspectors reviewed the documentation provided by the licensee and recognizes that the corrective actions applied in 2001 to the welds 3 and 4 served to reduce the tensile stresses in the welds, and although not documented in the generic implications section, the root cause report does address to an extent the justification for not considering these welds as currently susceptible to TGSCC. The corrective actions taken in 2001 included performing heat sink welding, which is a methodology used to reduce the stresses on the inner diameter (ID) of the weld, they also changed the design to reduce design stresses at weld #3 and they specified a smoother surface finish (RMS 125) to reduce potential crack initiation points.

However, the information provided did not demonstrate the compressive stresses on the ID will be retained during operations. Specifically, there were repairs conducted on the inner surface of weld #4, resulting in residual tensile stress at the inside surface of the weld which would promote the initiation of TGSCC.

Repairs were also performed on weld #3; these repairs were all conducted from the outer diameter (OD) surface of the weld. The licensee stated that due to this, the advantages of the heat sink welding process would be preserved. It is the inspectors position that an analysis of the actual weld and associated repair is necessary to determine the stress fields at the intersection of the new and existing welds in order to conclude the ID of the weld is unaffected by this process. The inspectors also noted the repairs were not performed using the heat sink welding process unless it consisted of removing the entire weld.

The inspectors determined the three factors required for TGSCC could still be present at the welds 3 and 4. These are:

- Corrosive environment – Weld #3 is in an environment essentially the same as the weld build up region. In the case of weld #4 though it is exposed to a lower temperature then the weld build up region, TGSCC could still occur at 250 degrees Fahrenheit as evidenced by previous cracking identified in the seal housings which are subject to lower temperatures.

- Susceptible material – Welds 3 and 4 are composed of the same material as the weld buildup region. This material also contains essentially the same material properties, with regards to susceptibility to TGSCC, as the pre-2001 CRDM housing design. This is the design that developed a through wall leak at weld #3.

- Tensile stresses above a certain unknown level - Fabrication and repairs will result in tensile stresses at the inside surface of weld #4 and possibly weld #3. The specific threshold for tensile stress that needs to be exceeded is unknown and the stresses in the weld are also unknown, especially those subject to grinding and re-weld. Therefore it is unknown whether the stresses in the weld are low enough to preclude TGSCC.

Though the root cause report states specifically the probable root cause of the cracking to be stresses in the weld buildup area due to manufacturing irregularities and misalignments between CRD-24 upper housing, support tube, seismic supports and the associated reactor head penetration/CRDM nozzle, it also states that based on the lack of cracking found in the other 8 upper housings tested, the failed CRD-24 upper housing contains an as-yet unidentified additional stress. Based on the potential existence of an additional stress, it cannot be concluded that this stress exists only in the weld build up region of CRDM housing #24. Therefore the inspectors do not agree that welds 3 and 4 should be excluded when evaluating the conditions for potential generic implications. Specifically, after reviewing the material provided by the licensee, the inspectors determined there is not sufficient evidence to exclude any of the three key elements needed to prevent TGSCC.

<u>Analysis:</u> The inspectors determined that the failure to adequately evaluate the generic implications of the cause of the cracking identified in CRDM #24 in accordance with the root cause procedure was a performance deficiency that warranted a significance evaluation. Specifically the licensee did not provide adequate justification for why no additional corrective actions associated with welds 3 and 4 was warranted, or take corrective action as necessary to address the potential generic implication. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. The inspectors also answered "yes" to the More-than-Minor screening questions "if left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, the licensee did not limit the likelihood of events that upset plant stability by not adequately evaluating the potential generic implications associated with welds 3 and 4, which could potentially result in another through wall

leak. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

In accordance with Table 3 "SDP Appendix Router" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors answered "no" to all the questions in Sections A through E, and were directed to IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power." The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function.

The inspectors determined that the primary cause of the failure to adequately consider welds 3 and 4 on the generic implications section of the root cause report related to the cross-cutting component of Human Performance, Decision Making, because licensee staff did not use conservative assumptions in decision making. Specifically, the licensee did not use conservative assumptions when excluding welds 3 and 4 as being susceptible to TGSCC and therefore include them in the generic implications section of the root cause report. (Item H.1(b) of IMC 310).

Enforcement: The inspectors identified a NCV of 10 CFR Appendix B Criterion V "Instructions, Procedures and Drawings", having a very low safety significance (Green), for failure to adequately evaluate the generic implications of the cause of cracking identified in CRDM #24 as it relates to weld 3 and 4 in accordance with the root cause procedure.

Title 10 CFR Appendix B Criterion V "Instruction, Procedures and Drawings requires in part, "Activities affecting quality shall be prescribed by documented procedures and shall be accomplished in accordance with these procedures."

Procedure EN-LI-118 Root Cause evaluation process states:

a. 5.5 (12)e: perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affects other SSC's , organizations or work processes. Use the two step process in accordance with attachment 9.7
b. Attachment 9.7 states Determine whether the occurrence/consequence (problem) is isolated, or whether it has broader (generic or common mode) implications. Achieve this by asking the following questions:
    i. Could this happen to equipment that is similar in function, design, or service condition?
    ii. Could this happen to a group of components? (components of the same construction or materials that could be similarly affected by one condition)

c. Attachment 9.7 also states: Document the results of the above considerations. Include the following items in the write up:
    i. Generic Implications (Is this problem/ cause limited to this component/equipment, or does it apply to others as well)
    ii. Existing broader (generic/common mode) considerations
d. 5.5(15)(10)c&f: Document proposed corrective actions and due dates to address valid generic implications. If no corrective action is recommended for a valid generic implication then document the basis for this conclusion and any risk or consequence identified as a result of taking no action.

Contrary to the above, as of March 15, 2013, the licensee failed to perform an activity affecting quality in accordance with prescribed procedures. Specifically, the licensee did not fully consider the generic implication of the cause of cracking in CRDm #24 by not adequately evaluating the root and contributing causes for their affects on welds 3 and 4. Therefore the licensee did not develop corrective actions to address these welds or provide an adequate basis for not taking any corrective actions related to the welds as required by the root cause procedure. Because of the very low safety significance and because the licensee entered this issue into their corrective action program (CR-PLP-2013-01500), it is being treated as a NCV consistent with Section 2.3.2 of the Enforcement Policy (NCV 05000255/2013003-xx).

4OA5 Other Activities

.1 (Closed) Unresolved Item 05000255/2012012-02: Potential Inadequate Degradation Evaluation of CRDM Housings

During a Special Inspection performed in August 2012, NRC inspectors identified an issue which could not be resolved without additional information (Unresolved Issue (URI)). This issue was associated with the rate of growth of the crack which created the through wall leak in CRDM housing 24, discovered on August ** 2012. Identification of this crack growth rate is significant in determining appropriate intervals for future inspections to provide reasonable assurance that CRDM housing leakage will not recur.

Preliminary failure analysis data available at the time of the inspection indicated that the observed cracking was due to transgranular stress corrosion cracking. Cracking of this type is normally due to the presence of oxygen and chlorides at the location of the crack. When examining the fracture surface at the location the through-wall leak occurred, the licensee identified six concentric rings (beach marks) propagating in a radial direction from the inside diameter out towards the outside diameter of the housing. Beach marks are normally associated with fatigue failures and indicate the number of stress cycles from crack initiation to crack failure. In this case there was no evidence that fatigue contributed to the failure. Despite the lack of evidence of fatigue, it was apparent that the crack which resulted in the CRDM housing 24 leak grew in increments. It was not, however, immediately apparent whether the increments were related to oxygen ingress (refueling outages) or temperature/pressure cycles.

At the time of the original inspection, 5 time intervals for through wall crack growth were under consideration. Two were based on literature crack growth data and three were based on interpretations of the beach marks. These time intervals were:

1. Based on literature data, one contractor estimated that a 10% through wall flaw would require 4 years to reach 50% through wall.
2. Based on literature data another contractor estimated the crack growth rate to be 2.1 x 10^{-5} in/hr or 0.18 in/yr. This is approximately three times faster than the crack growth rate proposed in the above mentioned rate.
3. Based on the concept of oxygen ingress at refueling outages 6 cycles of 18 months duration would require 9 years for the crack to grow through wall
4. Based on the concept of temperature/pressure cycles, the plant experienced 6 cold shutdowns in approximately 2 years preceding the crack. This equates to 2 years for the crack to grow through wall.
5. Based on the concept that oxygen is required for crack growth and that oxygen is rapidly purged from the CRDM housings due to leakage past the seals, crack growth occurs only during the first few weeks of operation following a refueling outage, followed by no growth for the remaining period of operation when oxygen concentrations are low. This equates to 6 oxygen ingress events (irrespective of time between events) for the crack to grow through wall.

NRC inspectors including technical experts from NRC Headquarters performed a follow-up inspection to determine if the assumptions made by the licensee were conservative and the planned actions bounded those conservative assumptions. The inspectors reviewed a variety of documents associated with crack growth and inspection intervals. The inspectors noted various the following statements included in the root cause report and vendor documents related to the determination of the appropriate crack growth rate.

1. The laboratory conducting the failure analysis concluded, it could not be conclusively determined if the beach marks corresponded to refueling outages, (i.e., 18 month cycle) or shorter periods as occurred during outages over the past 24 months
2. Palisades CRDM housing 21 leaked at weld 3 in 2001. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 failure. In calculating the crack growth rate of this crack, Contractor B utilized an interval between beach marks which is much shorter than refueling outages. The intervals used are consistent with plant thermal cycles in which oxygen may or may not have been admitted into the CRDMs.
3. A spare CRDM housing at Ft Calhoun leaked at weld 5 in 1990. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 failure. In calculating the crack growth rate of this crack, Ft Calhoun stated that the beach marks were related to refueling cycles. Ft Calhoun also performed calculations indicating that the oxygen level at the location of the flaw did not change with time (including in response to refueling outages) because the spare CRDM housing was not vented. Ft Calhoun's evaluation indicated that oxygen levels at the vicinity of the crack would have begun to decline through diffusion and convection had the intervals between outages been much longer than 18 months. This is interpreted to mean that the beach marks at Ft Calhoun are in response to pressure/thermal cycles.
4. In at least one instance Palisades needed to repair the seals on a reactor coolant pump at a time other than an outage. This necessitates draining some of the water

from the reactor coolant system and venting (admitting oxygen into) the CRDM housing. This represents an additional oxygen ingress event not included when determination of time to cracking is based on refueling outages.

5. In its inspection plan Palisades states that it will inspect all CRDM housings over the next 4 refueling outages, i.e., the interval between inspections is 1 refueling outage

Based on the above review, the inspection team notes that there are certain non conservative statements contained in the Root Cause Report and the inspection plan. These include:

1. The crack growth rate based on refueling outages is understated. If oxygen ingress is related to beach marks, given the oxygen ingress event which occurred to repair reactor coolant pump seals, 6 beach marks would occur in a maximum of 5 refueling intervals rather than the 6 refueling intervals that were used to calculate the crack growth rate in the root cause report.
2. The crack growth rate based on heat up and cool down cycles is overstated. The value in the root cause is based on 11 months. While 6 shutdowns did occur at the plant in 11 months several of these events did not result in pressure/temperature changes of the reactor coolant system. The appropriate time frame is 24 months rather than 11.
3. The inspection plan contains a non conservative statement: "However, once the crack has been initiated it propagates over 4 to 5 operating cycles prior to going through wall" While this statement does reflect one of the proposed theories for crack growth, sufficient evidence to demonstrate reasonable assurance that this theory is correct, and thereby overcome the non-conservatism of this statement, does not exist.

Despite the existence of the non conservatisms stated above, the inspectors conclude:

1. Sufficient evidence to conclusively determine the rate of crack growth does, and will not exist.
2. Crack growth based on pressure/temperature cycles is the most conservative of the potential crack growth mechanisms. In the absence of reasonable assurance of the correctness of less conservative mechanisms, through wall crack growth in 2 years must be utilized for regulatory purposes.
3. The licensee has not formally committed to any of the crack growth mechanisms discussed.
4. The licensee's inspection program includes inspections in each of the next 4 outages. This inspection interval, once per outage, bounds all the crack growth mechanisms considered.

The staff finds this approach to inspection to be both acceptable and sufficient justification to close this URI.

4OA6 Management Meetings

.2 Interim Exit Meetings

An interim exit was conducted for:

- The results of the selected issue follow-up inspection, with Mr. C. Arnone, Nuclear Safety Assurance Director on April 18, 2013.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

B. Davis, Engineering Director
O. Gustafson, Licensing Manager
T. Foudy, Engineering Supervisor
B. Williams, Engineer
B. Dotson, Licensing

LIST OF ITEMS OPENED, CLOSED, DISCUSSED

Closed

05000255/2012012-01	URI	TS for PCS Pressure Boundary Leakage
05000255/2012012-02	URI	Potential Inadequate Degradation Evaluation of CRDM Housings
05000255/2012012-03	URI	Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality

Opened and Discussed

None.

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

4OA5 Other Activities

LIST OF ACRONYMS USED

Sanchez Santiago, Elba

From: Holmberg, Mel
Sent: Tuesday, April 16, 2013 8:32 AM
To: Sanchez Santiago, Elba; Alley, David
Subject: RE: Palisades inspection report input
Attachments: Palisades Input to DRP Report 2013 002 URI EMS -msh comments.docx

Elba and Dave,

I have done some noodling with the proposed report draft (Elba, I did more work on your first finding from my input yesterday so please look at this version). Also, please make sure I did not introduce anything that is not true!!! It appears we may have a cross cutting aspect on the first finding, so Elba should evaluate this and discuss with SR resident inspector Tom. Elba I recommend you read from your latest draft of your input at your phone exit meeting Thursday especially if you add a cross-cutting aspect to finding No. 1.

M

From: Sanchez Santiago, Elba
Sent: Monday, April 15, 2013 12:48 PM
To: Holmberg, Mel
Subject: Palisades inspection report input

Mel,

I attached the draft palisades inspection report input in case you have a chance to review and provide me your input.

Thanks,

Elba M. Sanchez Santiago
Reactor Engineer
RIII/ DRS/ EB1
630-829-9715

C24

UNITED STATES
NUCLEAR REGULATORY COMMISSION
LISLE, IL 60532-4352

April XX, 2012

MEMORANDUM TO: Thomas Taylor
Senior Resident Inspector
Palisades Nuclear Plant

FROM: David Hills, Chief
Engineering Branch 3
Division of Reactor Safety

SUBJECT: PALISADES NUCLEAR PLANT
DRS INPUT TO INTEGRATED REPORT 05000255/2013002

Enclosed is the report input for the Palisades Nuclear Plant, Inspection Report 05000255/2013002. This report input documents completion of our review of Unresolved Items 05000255/2012012-01, “TS for PCS Pressure Boundary Leakage,” 05000255/2012012-02, “Potential Inadequate Degradation Evaluation of CRDM Housings,” and 05000255/2012012-03, “Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality.” This report also completes the one sample of the Problem Identification and Resolution, Selected Issue Follow-up in accordance with IP 71152. I have reviewed this input to confirm compliance with Inspection Manual Chapter (IMC) 0612 and IMC 305. This input is ready for inclusion into the integrated report and dissemination to the public.

Please input the following post Inspection Data into RPS:

Inspection Procedure	**Procedure Status** – see below: Incomplete, Complete, Complete by reference, Complete-full sample not available, Complete – opportunity to apply procedure not available, Not Applicable.	**Sample Size** – As documented in Scope Section If less than full sample size documented in the report input, the inspector must provide a justification below to enter into RPS and support the procedure status selected
71152	Complete	1

Inspection Report Item and Type (AV, FIN, NCV, URI or VIO)	**Cornerstone** (IE, MS, BI, EP, OR, PR, MISC)	**Cross Cutting Aspect** (H.n(i), P.n(i), S.n(i))	**Responsible Person/Owner**	**Procedure or TI** (71111.07T)	**RPS Branch Code (e.g. closeout responsibility)** EB1 3820 EB2 3870 EB3 3840 PST (RP) 3860 PSB (Safeguards) 3850 OB 3810
NCV-XXX	IE	n/a	E. Sanchez Santiago	71152	3820
NCV-XXX	IE	H.1(b)	E. Sanchez Santiago	71152	3820

Enclosure: Input to Inspection Report 05000255/2013002

cc w/encl: J. Giessner, Chief
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DATE	4/ /13							

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Cover Letter

X Green findings involving a violation were identified. Include the following:

Based on the results of this inspection, two NRC-identified findings of very low safety significance (Green) were identified. These findings were determined to involve a violation of NRC requirements. However, because of the very low safety significance and because the issues were entered into your corrective action program, the NRC is treating the issue as Non-Cited Violation, in accordance with Section 2.3.2 of the NRC Enforcement Policy.

TITLE PAGE

Inspectors: D. Alley, Senior Materials Engineer
E. Sanchez Santiago, Reactor Inspector

SUMMARY OF FINDINGS

A. NRC-Identified and Self-Revealed Findings

Cornerstones: Initiating Events

- Green. The inspectors identified a Non-Cited Violation (NCV) of 10 CFR Part 50 Appendix B Criterion XVI, and Technical Specification (TS) 3.4.14 Primary Coolant System (PCS) Operational Leakage, for failure to prevent recurrence of a significant condition adverse to quality resulting a non-compliance with the TS. Specifically, the licensee failed to take adequate corrective actions in response to a pressure boundary leak from CRDM housing in 2001 which resulted in a pressure boundary leak from a similar CRDM housing in August 2012. The licensee operated with this pressure boundary leak for greater than the TS allowed time. The licensee entered this issue into their corrective action program as CR-PLP-2013-01134.

  The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. Specifically the licensee did not limit the likelihood of events that upset plant stability by not taking adequate corrective actions to prevent recurrence of leakage in CRDM housings which represents a pressure boundary leakage and a condition prohibited by the technical specifications. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the because the failure of a CRDM housing is a Primary System LOCA initiator contributor. In accordance with Table 3 "SDP Appendix Router" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors answered "no" to all the questions in Sections A through E, and were directed to IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power." The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The

Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors determined that finding was not indicative of current performance and therefore a cross-cutting aspect was not applied. (Section 4OA2.b(1))

- <u>Green.</u> The inspectors identified a Green Non-Cited Violation (NCV) of 10 CFR Part 50 Appendix B Criterion V, for failure to follow the root cause procedure. Specifically, the licensee failed to adequately evaluate the generic implications of the cause of the cracking identified in CRDM #24. The licensee entered this issue into their corrective action program as CR-PLP-2013-01500.

  The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. The inspectors also answered "yes" to the More-than-Minor screening questions "if left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, the licensee did not limit the likelihood of events that upset plant stability by not adequately evaluating the potential generic implications associated with welds 3 and 4, which could potentially result in another through wall leak. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the because the failure of a CRDM housing is a Primary System LOCA initiator contributor. In accordance with Table 3 "SDP Appendix Router" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors answered "no" to all the questions in Sections A through E, and were directed to IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power." The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors determined that the primary cause of the failure to adequately consider welds 3 and 4 on the generic implications section of the root cause report related to the cross-cutting component of Human Performance, Decision Making, because licensee staff did not use conservative assumptions in decision making. Specifically, the licensee did not use conservative assumptions when excluding welds 3 and 4 as being susceptible to TGSCC and therefore include them in the generic implications section of the root cause report. (Item H.1(b)). (Section 4OA2.b(2))

B. <u>Licensee-Identified Violations</u>

No violations of significance were identified.

REPORT DETAILS

4. REACTOR SAFETY

4OA2 Identification and Resolution of Problems (71152)

.3 Selected Issue Follow-up Inspection: Through Wall Leakage of Control Rod Drive Mechanism (CRDM) Housing #24

a. Inspection Scope

On August 12, 2012 the licensee shutdown to investigate an increase in unidentified leakage. The source of the leakage was determined to be a crack in control rod drive mechanism housing (CRDM) No. 24. Shortly after the discovery of the leak in CRDM housing No. 24, the NRC dispatched a special inspection team (SIT) to review the CRDM No. 24 leakage event. The SIT identified an unresolved item (URI) related to the potential failure to prevent recurrence of a significant condition adverse to quality (SCAQ) which was considered an unresolved item, because the licensee's root cause investigation was ongoing at that time. The licensee subsequently removed the failed housing from service for further testing and completed an evaluation to determine the cause of the cracking (reference root cause report xxx).

From xxx to yyy, the inspectors completed one inspection sample regarding problem identification and resolution based upon review of the licensee's root cause report xxx and associated corrective action records related to this issue:

- CR

The inspectors reviewed the licensee's actions in accordance with performance attributes identified in IP 71152. Specifically, the inspectors reviewed licensee corrective action records to determine if: (1) the problems were accurately identified; (2) operability and reportability were adequately ascertained; (3) extent of condition and generic implications were appropriately addressed; (4) classification and prioritization of problem was commensurate with safety significance; (5) root and contributing causes were identified; (6) corrective actions were appropriately focused to correct problem; and (7) timely corrective actions were completed or proposed commensurate with the safety significance of the issues.

~~As a follow-up to this issue the NRC performed an inspection of the actions taken by the licensee in response the CRDM housing through wall crack. The inspection consisted of a review of the root cause report as well as supporting documentation provided by vendors, such as calculations and technical evaluations. The inspectors also reviewed available operating experience related to this issue.~~

b. Findings

.1 Failure to Prevent Recurrence of a Significant Issue Adverse to Quality

Introduction: The inspectors identified a Green Non-Cited Violation (NCV) of 10 CFR Part 50 Appendix B Criterion XVI, and Technical Specification (TS) 3.4.14 Primary Coolant System (PCS) Operational Leakage, for failure to prevent recurrence of a significant condition adverse to quality resulting a non-compliance with the TS. Specifically, the licensee failed to include the internal CRDM weld build-up area within the scope of corrective actions taken for a 2001 CRDM housing leakage event (a significant condition adverse to quality) and consequently leakage recurred at the CRDM housing No. 24 in 2012.

~~take adequate corrective actions in response to a pressure boundary leak from CRDM housing in 2001 which resulted in a pressure boundary leak from a similar CRDM housing in August 2012. The licensee operated with this pressure boundary leak for greater than the TS allowed time.~~

Description: On August 12, 2012 Palisades Nuclear Power Station shutdown to investigate an increase in unidentified leakage. During a walk-down performed post shutdown the licensee discovered the source of the leakage to be a pressure boundary leak from ~~Control Rod Drive Mechanism~~ (CRDM) Housing No.#24. After further testing, the licensee ~~it was~~ determined the leak ~~was associated with~~ occurred because of through-wall flaw adjacent to a weld build up on the interior of the housing. The purpose of this weld build-up is to maintain the ~~control drive mechanism~~ CRDM properly aligned.

The licensee formed a root cause team (RCT) staffed with xxx site and yy vendor staff that conducted the root cause investigation in accordance with site procedures procedures xx and yy and on xx, issued a root cause analysis report zzz. In this report, the licensee's RCT ~~The licensee performed a root cause analysis to~~ determined that the probable cause of the cracking was "Stresses in the weld build up area due to manufacturing irregularities and misalignments between CRD-24 upper housing, support tube, and the associated reactor head penetration/CRDM nozzle. Based on lack of cracking found in the other 8 upper housings tested, the failed CRD-24 upper housing contains an as-yet unidentified additional stress". The ~~report~~ RCT also identified the following contributing cause: "Transgranular Stress Corrosion Cracking (TGSCC) initiating within the internal weld build-up material of CRD-24. The through wall crack initiated in the weld material and then propagated through the base metal until a leak developed in the OD witness band region at the base of the inner diameter (ID) weld build up. ~~The investigation consisted of performing~~ This conclusion was based upon ~~testing on the failed CRDM housing which included~~ destructive and non destructive examinations (NDE) ~~and as well as destructive analyses~~ completed on a section of the failed housing which included the through-wall flaw. The ~~licensee's~~ RCT ~~investigation~~ also ~~obtained vendor~~ relied upon vendor technical reports assessing ~~on~~ the results of the NDE ~~examinations~~ as well as vendor calculations related to the stresses in the CRDM housings.

~~In order~~ To determine the extent of condition, the licensee performed ultrasonic (UT) examinations ~~testing~~ of the weld build up area on 8 additional CRDM housings. ~~The results of the root cause analysis were documented in a root cause report which was reviewed by the NRC inspectors during the follow-up inspection. The root cause report defined the probable root cause as:~~ The licensee selected these locations because xxx. The inspectors concluded that this was an appropriate extent of condition review based upon the cause of the CRDM No. 24 failure identified by the licensee (or if we disagree need to understand why it is not an immediate safety issue).

~~An event similar to this occurred~~ In 2001 ~~when~~ the licensee discovered a steam leak in the housing of CRDM-21 caused by a through-wall TGSCC at CRDM housing weld No. 3 which was located just below the weld build-up region. ~~which was also classified as pressure boundary leakage. In this case the crack was associated with a butt weld located just below the aforementioned weld buildup.~~ This issue was categorized as a significant issue adverse to quality by the licensee (reference CR xx) and the licensee's root cause evaluation was documented in RCR/CR-xxx ~~performed~~. The ~~root cause evaluation~~ licensee concluded that the cracks in CRDM-21 were caused by TGSCC which occurred in areas of heavy grinding or machining tool marks. Specifically, this leak ~~in the housing~~ was the result of an inner diameter initiated, axially oriented, transgranular crack in the austenitic stainless steel housing material. The licensee's extent of condition investigation identified TGSCC at most of the inservice housings near the weld No. 3 location. The licensee's corrective actions ~~taken by the licensee~~ included replacing all 45 CRDM housings with modified housings. ~~when subsequent testing indicated additional cracks in the same location in other CRDM housings~~. The modifications that the licensee made to the replacement housings to prevent recurrence included: ~~controlling the fabrication process for the pressure retaining welds in the CRDM housing to ensure~~ prohibiting grinding ~~was not performed and the~~ at the ID surface during fabrication so that tensile residual stresses on the internal surface of weld were reduced; ~~The licensee also modified the physical~~ changing the ~~design~~ location of weld No. 3?? to reduce the design stresses and the accumulation of contaminants in proximity to the weld; ~~where the cracking had occurred. The licensee also~~ and changing the material of the housing from type 347 to type 316 stainless steel ~~component., though the replacement material was essentially equal to the previous material when comparing susceptibility to TGSCC. the actions specified by the licensee~~ to prevent recurrence of ~~cracking in the CRDM housings.~~

To evaluate the effectiveness of these preventative actions from the 2001 CRDM leakage event, the inspectors reviewed the licensee actions to determine if they had been sufficient to eliminate one of the 3 necessary factors to cause TGSCC on the CRDM housings: (1) a susceptible material, (2) a corrosive environment and (3) tensile stress." The inspectors identified that the licensee had failed to eliminate one or more of the necessary factors at the weld build-up area to preclude TGSCC in the replacement housing. Specifically:

- The licensee's 2001 root cause report documented that the weld build-up region is exposed to essentially the same environment as the weld that experienced the cracking (corrosive environment remained unchanged).

- No analysis was completed on the stress conditions for the weld build-up region prior to approving the modified replacement housing design (left residual tensile weld stresses on ID of CRDM surface).

- Fabrication restrictions to prohibit grinding were not applied to the weld build-up region (grinding promotes residual tensile stress state on ID of CRDM surface)
- Cold work (??? What specific type of cold work??)was applied to weld buildup areas during fabrication (induced residual tensile stresses)
- Material was changed from type 347 to type 316 stainless steel (both materials are essentially equally susceptible to TGSCC).

Based upon the recurrence of through-wall leakage in the CRDM housings that occurred at the weld buildup region of the CRDM housings by TGSCC, the inspectors concluded that the licensee actions were not adequate because the appropriate actions to preclude recurrence were within the licensee's ability to foresee and implement. In 1991, the Fort Calhoun plant had experienced through-wall leakage due to TGSCC at the weld build-up region of their CRDM housings (same housing design) and this operational experience had been reviewed by the licensee and dismissed. In CR xxx, the licensee documented their review of the weld build-up region failure by TGSCC at Fort Calhoun in the spare housing and concluded it would not occur at Palisades. This conclusion was based on the assumption that a higher oxygen environment (more aggressive environment) would exist in the spare Fort Calhoun housings than in the inservice Palisades housings. However the licensee did not have a sufficient basis to confirm this assumption, nor did the licensee perform additional testing to determine if the environment of their inservice housings was sufficiently benign to prevent TGSCC. The licensee's 2012 RCT also reached a similar conclusion and documented that due to organizational/ programmatic weakness ~~section of the 2012 root cause evaluation the licensee states~~ the 1991 Fort Calhoun operating experience was not adequately utilized to include inspection of the housing ID weld build-up regions. The inspectors identified that the licensee had an missed a key opportunity to implement effective corrective actions that could have prevented recurrence of the 2001 leakage event and elected not to pursue these actions because of the cost. Specifically, in CR xxx the licensee considered fabricating the replacement housings with Inconel 600 material because it was much more resistant to TGSCC. However, the licensee elected not to fabricate the replacement housings using this material because of the increased cost.

~~Through their review of the 2001 root cause report, the inspectors noted the corrective actions were focused on the pressure retaining welds contained in the CRDM housing and due to various reasons the weld build up region was excluded from the analyses and technical assessments performed in response to the through wall leak. The licensee also considered operating experience which included an incident at Fort Calhoun where a through-wall crack had developed in the weld build-up region of their CRDM housings. Fort Calhoun is the only additional plant to have the same CRDM housing design as Palisades. When making the comparison the licensee discounted the weld build up region because it did not meet the exact characteristics of the Fort Calhoun incident.~~

~~The inspectors also had various discussions with the licensee to address questions and concerns related to this issue. The activities performed by the inspectors also included internal discussions with regional inspectors and supervisors as well as technical experts from headquarters. Through their review of the information available and the internal and external discussions,~~

During the special inspection the inspectors also identified an unresolved item for the Technical Specification pressure boundary leak. The licensee determined the leakage commenced on July 14, 2012. The licensee operated in this condition for greater than 6 hours, which is the required shutdown time when pressure boundary leakage exists in the plant. Based on the information provided above, unresolved items 05000255/2012012-01 "TS for PCS Pressure Boundary Leakage" and 05000255/2012012-03 "Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality," are being closed to the following finding and associated violation.

Analysis: The inspectors determined that the licensee's failure to prevent recurrence of TGSCC of the CRDM housings (a significance condition adverse to quality) that resulted in a TS non-compliance was a performance deficiency that warranted a significance evaluation. ~~Specifically, the licensee failed to recognize the susceptibility of the weld build up region to TGSCC and therefore did not apply the level of scrutiny and corrective actions to this weld resulting in a failure to prevent recurrence of leakage in the CRDM housing due to TGSCC.~~ The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. Specifically the licensee did not limit the likelihood of events that upset plant stability by not taking adequate corrective actions to prevent recurrence of leakage in CRDM housings which represents a pressure boundary leakage and a condition prohibited by the technical specifications. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

~~In accordance with Table 3 "SDP Appendix Router" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors answered "no" to all the questions in Sections A through E, and were directed to IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power."~~ The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a large component rupture.

The inspectors determined that finding is not indicative of current performance and therefore a cross cutting aspect is not applied. (why not ??? explain. Based on finding below it looks like they made the same mistakes with the current corrective actions to prevent recurrence)

Enforcement: The inspectors identified a NCV of 10 CFR Appendix B Criterion XVI "Corrective Actions", and Technical Specification 3.4.13 "Primary Coolant System Operational Leakage", having a very low safety significance (Green), for failure to prevent the recurrence of leakage in CRDM housings due to TGSCC resulting in the operation of the reactor with pressure boundary leakage for greater than the TS allowed time.

Title 10 CFR Appendix B Criterion XVI requires, in part, that "In the case of significant conditions adverse to quality, the cause of the condition is determined and corrective action taken to preclude repetition."

Technical Specification 3.4.13 PCS Operation Leakage states, in part, "PCS operational Leakage shall be limited to no pressure boundary leakage." Condition B requires the licensee be in Hot Standby in 6 hours and Cold Shutdown in 36 hours when pressure boundary leakage exists.

Contrary to the above, from June 2001 through the end date for their last corrective action for the 2001 event (or if not clear can use all the way to exit date April 17, 2012), the licensee failed to take adequate corrective actions to prevent recurrence of pressure boundary leakage in CRDM housings due to TGSCC. The leakage of CRDM housing # 21 identified in 2001 was categorized as a significant condition adverse to quality in accordance with the licensee's corrective action program. The licensee performed a root cause evaluation that determined the a contributing (need to check with EICS to see if it matters that this was not the root cause???) cause to be TGSCC. The corrective actions to prevent recurrence included changing the design to reduce stresses in the failed weld, control the surface finish of the pressure retaining welds to reduce potential crack initiation points and the welding process was also changed to reduce the stresses in the weld. These corrective actions were narrowly focused on the pressure retaining welds of the CRDM housings. As a result on August 12, 2012 a leak was identified from CRDM housing #24. The cause of this leak was also determined to be TGSCC. The source of

the leakage was specifically the weld build up region, which was ~~inadequately~~ inappropriately excluded from the scope of corrective actions taken in 2001 to prevent recurrence. The pressure boundary leakage was identified due to an increase in unidentified leakage noted on July 14, 2012. The plant did not enter Hot Standby until August 12, 2012 indicating the licensee operated with pressure boundary leakage for greater than the TS allowed time of 6 hours. (Need to identify actions proposed to restore compliance with criterion XVI???)
Because of the very low safety significance and because the licensee entered this issue into their corrective action program (CR-PLP-20136-01134), it is being treated as a NCV consistent with Section 2.3.2 of the Enforcement Policy (NCV 05000255/2013002-xx).

.2 Failure to Adequately Address the Generic Implications of the Cracking identified in CRDM 24

Introduction: The inspectors identified a Green Non-Cited Violation (NCV) of 10 CFR Part 50 Appendix B Criterion V, for failure to follow the root cause procedure. Specifically, the licensee failed to adequately evaluate the generic implications of the cause of the cracking identified in CRDM No. 24.

Description: While reviewing the 2012 root cause report xxx related the cracking identified in CRDM #24 the inspectors identified that the licensee had not appropriately considered the ~~a concern related to the~~ generic implications of the cracking in the extent of condition review. The licensee's proposed corrective actions ~~going forward~~ were ~~solely~~ narrowly focused on the weld build up region, instead of broader actions to ensure other CRDM housing welds were fit for their intended service life. ~~which is where the current failure occurred.~~

On March xxx, the inspectors requested that the licensee identify the bases for excluding other CRDM housing welds (weld #3 below the weld build up region and weld #4 above the weld build up region) from the scope of planned corrective actions. On xxx, the licensee provided additional information to justify excluding these welds from the scope of the corrective actions. The licensee stated "xxx." The licensee credited the corrective actions associated with the modifications to the CRDM housing design completed in 2001 as the basis to exclude housing welds No 3 and 4 from additional actions to identify the extent of TGSCC. The corrective actions taken in 2001 included performing heat sink welding, which is a methodology used to reduce the stresses on the inner diameter (ID) of the weld, they also changed the design to reduce design stresses at weld #3 and they specified a smoother surface finish (RMS 125) to reduce potential crack initiation points. The inspectors acknowledged that these actions would reduce the tensile stress at the ID surface and thus reduce the probability of initiating TGSCC. ~~NRC inspectors reviewed the documentation provided by the licensee and recognizes that the corrective actions applied in 2001 to the welds 3 and 4 served to reduce the tensile stresses in the welds, and although not documented in the generic implications section, the root cause report does address to an extent the justification for not considering these welds as currently susceptible to TGSCC.~~

However, the information provided did not demonstrate that TGSCC would not occur because it did not demonstrate that tensile stress would be eliminated at the

~~compressive stresses on the~~ ID surface ~~will be retained~~ during operation. In particular, ~~Specifically, there were~~ repairs completed ~~conducted on~~ at the inner surface of weld No. 4, would result in high ~~resulting in~~ residual tensile stress at the inside surface of the weld which would promote the initiation of TGSCC. Repairs were also performed on weld No. 3; ~~these repairs were all conducted~~ from the outer diameter (OD) surface of the weld. The licensee believed that ~~stated that due to this, the advantages of~~ the last pass heat sink welding process would be sufficient to ensure residual compressive stress would remain at the ID surface of Weld No. 3 even with repairs to the OD surface. ~~preserved~~. However, the licensee had not completed detailed residual weld stress testing or modeling to confirm this assumption. ~~It is the inspectors position that an analysis of the actual weld and associated repair is necessary to determine the stress fields at the intersection of the new and existing welds in order to conclude the ID of the weld is unaffected by this process. The inspectors also noted the repairs were not performed using the heat sink welding process unless it consisted of removing the entire weld.~~

The inspectors identified that ~~determined~~ the three factors required for TGSCC could still be present at the welds 3 and 4 as follows:. ~~These are:~~

- Corrosive environment – Weld #3 would operate in ~~an~~ a similar environment ~~essentially the same~~ as the weld build up region of the CRDM housing. ~~In the case of~~ weld No. 4 ~~though it is exposed to~~ would be exposed to a lower operating temperature then the weld build up region, however TGSCC ~~could~~ can still occur at 250 degrees Fahrenheit as evidenced by the Palisades previous operating experience with cracking identified in the seal housings that operate at even lower temperatures. ~~which are subject to lower temperatures.~~

- Susceptible material – Welds 3 and 4 are composed of the same weld filler and base metal materials as the weld buildup region (e.g. weld filler material consistent with the type 316 stainless housing base metal). This material ~~also contains essentially the same material properties~~, ~~with regards to~~ would be equally susceptible to TGSCC, as the type 347 stainless steel and weld filler materials used in the pre-2001 CRDM housing design~~. This is the design~~ that developed a through wall leak caused byTGSCC at weld No.3.

- Tensile stresses above a certain unknown level - Fabrication and repairs will result in tensile stresses at the inside surface of weld #4 and possibly weld #3. The specific threshold for tensile stress that needs to be exceeded is unknown and the stresses in the weld are also unknown, especially those subject to grinding and re-weld. Therefore it is unknown whether the stresses in the weld are low enough to preclude TGSCC.

The licensee documented in the RCR ~~Though the root cause report states specifically~~ "the probable root cause of the CRDM cracking to be stresses in the weld buildup area due to manufacturing irregularities and misalignments between CRD-24 upper housing, support tube, seismic supports and the associated reactor head penetration/CRDM nozzle." Additionally the licensee stated~~, it also states~~ "that based on the lack of cracking found in the other 8 upper housings tested, the failed CRD-24 upper housing contains an as-yet unidentified additional stress." ~~Based on the potential existence of an additional stress, it cannot be concluded that this stress exists only in the weld build up region of CRDM housing #24. Therefore the inspectors do not agree that~~ Because the cause of the additional stress was not identified, the licensee had not established a sufficient basis in the RCR to exclude welds 3 and 4 ~~should be excluded when evaluating the conditions for~~ from the extent of condition review (e.g. potential generic implications).

The inspectors identified that the licensee had not followed Procedure EN-LI-118 Root Cause evaluation in the root cause review of the CRDM housing No. 24 leak as documented in report xxx. Section 5.5 (12)e of EN-LI-118 required that the licensee "perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affects other SSC's." In this case, the inspectors identified that the licensee had not documented a sufficient basis in RCR xxx to exclude welds No. 3 and No. 4 from the generic factors discussed above that led to the 2012 leak in the CRDM housing No. 24 (e.g. TGSCC at the weld buildup region). The licensee entered this issue into the corrective action program as AR yyy. To restore compliance with the procedure, the licensee intended to revise the root cause report xxx to add additional the appropriate generic evaluation and was considering additional corrective actions to inspect a sample of welds No. 3 and 4 for TGSCC during the upcoming refueling outage.

~~Specifically, after reviewing the material provided by the licensee, the inspectors determined there is not sufficient evidence to exclude any of the three key elements needed to prevent TGSCC.~~

<u>Analysis:</u> The inspectors determined that the failure to adequately evaluate the generic implications of the cause of the cracking identified in CRDM #24 in accordance with the root cause procedure EN-LI-118 was a performance deficiency that warranted a significance evaluation. ~~Specifically the licensee did not provide adequate justification for why no additional corrective actions associated with welds 3 and 4 was warranted, or take corrective action as necessary to address the potential generic implication.~~ The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. The inspectors also answered "yes" to the More-than-Minor screening questions "if left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, absent NRC identification, the licensee would not have completed further evaluations or inspections of CRDM housing welds which could have resulted in additional CRDM housing failure and leakage by TGSCC. ~~the licensee did not limit the likelihood of events that upset plant stability by not adequately evaluating the potential generic implications associated with welds 3 and 4, which could potentially result in another through wall leak.~~ In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

~~In accordance with Table 3 "SDP Appendix Router" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors answered "no" to all the questions in Sections A through E, and were directed to IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At Power."~~ The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June

19, 2012. Specifically, the inspectors answered “no” to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and “no” to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a large component rupture.

The inspectors determined that the primary cause of the failure to adequately consider welds 3 and 4 on the generic implications section of the root cause report related to the cross-cutting component of Human Performance, Decision Making, because licensee staff did not use conservative assumptions in decision making. Specifically, the licensee did not use conservative assumptions when excluding welds 3 and 4 as being susceptible to TGSCC and therefore include them in the generic implications section of the root cause report. (Item H.1(b) of IMC 310).

Enforcement: The inspectors identified a NCV of 10 CFR Appendix B Criterion V “Instructions, Procedures and Drawings”, having a very low safety significance (Green), for failure to adequately evaluate the generic implications of the cause of cracking identified in CRDM #24 as it relates to weld 3 and 4 in accordance with the root cause procedure.

Title 10 CFR Appendix B Criterion V “Instruction, Procedures and Drawings requires in part, “Activities affecting quality shall be prescribed by documented procedures and shall be accomplished in accordance with these procedures.”

Procedure EN-LI-118 Root Cause evaluation process (revision xxx) states:

a. 5.5 (12)e: perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affects other SSC’s , organizations or work processes. Use the two step process in accordance with attachment 9.7
b. Attachment 9.7 states Determine whether the occurrence/consequence (problem) is isolated, or whether it has broader (generic or common mode) implications. Achieve this by asking the following questions:
    i. Could this happen to equipment that is similar in function, design, or service condition?
    ii. Could this happen to a group of components? (components of the same construction or materials that could be similarly affected by one condition)
c. Attachment 9.7 also states: Document the results of the above considerations. Include the following items in the write up:
    i. Generic Implications (Is this problem/ cause limited to this component/equipment, or does it apply to others as well)
    ii. Existing broader (generic/common mode) considerations
d. 5.5(15)(10)c&f: Document proposed corrective actions and due dates to address valid generic implications. If no corrective action is recommended for a valid generic implication then document the basis for this conclusion and any risk or consequence identified as a result of taking no action.

Contrary to the above, From xxx (date root cause report was issued) through April 17, 2013 (date of exit meeting), the licensee failed to perform an activity affecting quality in accordance with ~~prescribed~~ procedure EN-LI-118. Specifically, the licensee did not identify and document the existing broader (generic/common mode) considerations associated with TGSCC at CRDM housing welds No. 3 and No. 4. ~~fully consider the generic implication of the cause of cracking in CRDm #24 by not adequately evaluating the root and contributing causes for their affects on welds 3 and 4.~~ Consequently, the licensee failed to propose corrective actions for the generic implications of TGSCC at CRDM housing welds No. 3 and No. 4. ~~Therefore, the licensee did not develop corrective actions to address these welds or provide an adequate basis for not taking any corrective actions related to the welds as required by the root cause procedure.~~ The licensee was considering xxx and yyy to restore compliance with the procedure EN-LI-118. Because of the very low safety significance and because the licensee entered this issue into their corrective action program (CR-PLP-2013-01500), it is being treated as a NCV consistent with Section 2.3.2 of the Enforcement Policy (NCV 05000255/2013003-xx).

4OA5 Other Activities

.1 (Closed) Unresolved Item 05000255/2012012-02: Potential Inadequate Degradation Evaluation of CRDM Housings

During a Special Inspection performed in August 2012, NRC inspectors identified an issue which could not be resolved without additional information (Unresolved Issue (URI)). This issue was associated with the rate of growth of the crack which created the through wall leak in CRDM housing 24, discovered on August ** 2012. Identification of this crack growth rate is significant in determining appropriate intervals for future inspections to provide reasonable assurance that CRDM housing leakage will not recur.

Preliminary failure analysis data available at the time of the inspection indicated that the observed cracking was due to transgranular stress corrosion cracking. Cracking of this type is normally due to the presence of oxygen and chlorides at the location of the crack. When examining the fracture surface at the location the through-wall leak occurred, the licensee identified six concentric rings (beach marks) propagating in a radial direction from the inside diameter out towards the outside diameter of the housing. Beach marks are normally associated with fatigue failures and indicate the number of stress cycles from crack initiation to crack failure. In this case there was no evidence that fatigue contributed to the failure. Despite the lack of evidence of fatigue, it was apparent that the crack which resulted in the CRDM housing 24 leak grew in increments. It was not, however, immediately apparent whether the increments were related to oxygen ingress (refueling outages) or temperature/pressure cycles.

At the time of the original inspection, 5 time intervals for through wall crack growth were under consideration. Two were based on literature crack growth data and three were based on interpretations of the beach marks. These time intervals were:

1. Based on literature data, one contractor estimated that a 10% through wall flaw would require 4 years to reach 50% through wall.
2. Based on literature data another contractor estimated the crack growth rate to be 2.1 x 10^{-5} in/hr or 0.18 in/yr. This is approximately three times faster than the crack growth rate proposed in the above mentioned rate.
3. Based on the concept of oxygen ingress at refueling outages 6 cycles of 18 months duration would require 9 years for the crack to grow through wall
4. Based on the concept of temperature/pressure cycles, the plant experienced 6 cold shutdowns in approximately 2 years preceding the crack. This equates to 2 years for the crack to grow through wall.
5. Based on the concept that oxygen is required for crack growth and that oxygen is rapidly purged from the CRDM housings due to leakage past the seals, crack growth occurs only during the first few weeks of operation following a refueling outage, followed by no growth for the remaining period of operation when oxygen concentrations are low. This equates to 6 oxygen ingress events (irrespective of time between events) for the crack to grow through wall.

NRC inspectors including technical experts from NRC Headquarters performed a follow-up inspection to determine if the assumptions made by the licensee were conservative and the planned actions bounded those conservative assumptions. The inspectors reviewed a variety of documents associated with crack growth and inspection intervals. The inspectors noted various the following statements included in the root cause report and vendor documents related to the determination of the appropriate crack growth rate.

1. The laboratory conducting the failure analysis concluded, it could not be conclusively determined if the beach marks corresponded to refueling outages, (i.e., 18 month cycle) or shorter periods as occurred during outages over the past 24 months
2. Palisades CRDM housing 21 leaked at weld 3 in 2001. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 failure. In calculating the crack growth rate of this crack, Contractor B utilized an interval between beach marks which is much shorter than refueling outages. The intervals used are consistent with plant thermal cycles in which oxygen may or may not have been admitted into the CRDMs.
3. A spare CRDM housing at Ft Calhoun leaked at weld 5 in 1990. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 failure. In calculating the crack growth rate of this crack, Ft Calhoun stated that the beach marks were related to refueling cycles. Ft Calhoun also performed calculations indicating that the oxygen level at the location of the flaw did not change with time (including in response to refueling outages) because the spare CRDM housing was not vented. Ft Calhoun's evaluation indicated that oxygen levels at the vicinity of the crack would have begun to decline through diffusion and convection had the intervals between outages been much longer than 18 months. This is interpreted to mean that the beach marks at Ft Calhoun are in response to pressure/thermal cycles.
4. In at least one instance Palisades needed to repair the seals on a reactor coolant pump at a time other than an outage. This necessitates draining some of the water from the reactor coolant system and venting (admitting oxygen into) the CRDM housing. This represents an additional oxygen ingress event not included when determination of time to cracking is based on refueling outages.
5. In its inspection plan Palisades states that it will inspect all CRDM housings over the next 4 refueling outages, i.e., the interval between inspections is 1 refueling outage

Based on the above review, the inspection team notes that there are certain non conservative statements contained in the Root Cause Report and the inspection plan. These include:

1. The crack growth rate based on refueling outages is understated. If oxygen ingress is related to beach marks, given the oxygen ingress event which occurred to repair reactor coolant pump seals, 6 beach marks would occur in a maximum of 5 refueling intervals rather than the 6 refueling intervals that were used to calculate the crack growth rate in the root cause report.
2. The crack growth rate based on heat up and cool down cycles is overstated. The value in the root cause is based on 11 months. While 6 shutdowns did occur at the plant in 11 months several of these events did not result in pressure/temperature changes of the reactor coolant system. The appropriate time frame is 24 months rather than 11.
3. The inspection plan contains a non conservative statement: "However, once the crack has been initiated it propagates over 4 to 5 operating cycles prior to going through wall" While this statement does reflect one of the proposed theories for crack growth, sufficient evidence to demonstrate reasonable assurance that this theory is correct, and thereby overcome the non-conservatism of this statement, does not exist.

Despite the existence of the non conservatisms stated above, the inspectors conclude:

1. Sufficient evidence to conclusively determine the rate of crack growth does, and will not exist.
2. Crack growth based on pressure/temperature cycles is the most conservative of the potential crack growth mechanisms. In the absence of reasonable assurance of the correctness of less conservative mechanisms, through wall crack growth in 2 years must be utilized for regulatory purposes.
3. The licensee has not formally committed to any of the crack growth mechanisms discussed.
4. The licensee's inspection program includes inspections in each of the next 4 outages. This inspection interval, once per outage, bounds all the crack growth mechanisms considered.

The staff finds this approach to inspection to be both acceptable and sufficient justification to close this URI.

4OA6 Management Meetings

.2 Interim Exit Meetings

An interim exit was conducted for:

- The results of the selected issue follow-up inspection, with Mr. C. Arnone, Nuclear Safety Assurance Director on April 18, 2013.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

B. Davis, Engineering Director
O. Gustafson, Licensing Manager
T. Foudy, Engineering Supervisor
B. Williams, Engineer
B. Dotson, Licensing

LIST OF ITEMS OPENED, CLOSED, DISCUSSED

Closed

05000255/2012012-01	URI	TS for PCS Pressure Boundary Leakage
05000255/2012012-02	URI	Potential Inadequate Degradation Evaluation of CRDM Housings
05000255/2012012-03	URI	Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality

Opened and Discussed

None.

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

4OA5 Other Activities

LIST OF ACRONYMS USED

Holmberg, Mel

From:	Sanchez Santiago, Elba
Sent:	Wednesday, April 17, 2013 9:04 AM
To:	Hills, David; Holmberg, Mel; Giessner, John
Cc:	Betancourt, Diana
Subject:	FW: Palisades CRD Upper Housing: Welds 3 and 4 White Paper
Attachments:	Weld 3 & 4 Justification rev 2.doc

FYI

From: Williams, Benjamin [mailto:bwill17@entergy.com]
Sent: Wednesday, April 17, 2013 8:01 AM
To: Sanchez Santiago, Elba; Alley, David; Taylor, Thomas; Scarbeary, April
Cc: Davis, Barry; Haumersen, Johannes; FOUTY, THOMAS H; GUSTAFSON, OTTO W; DAVIS, TERRY A; ERICKSON, JEFFREY S; DOTSON, BARBARA E
Subject: Palisades CRD Upper Housing: Welds 3 and 4 White Paper

Dear NRC and NRR Inspection Team,

Attached is a revised version of the weld 3 and 4 white paper for your consideration. It is Palisades' belief that the inspection of welds 3 and 4 is an enhancement.

Respectfully,

Ben Williams
System Engineering
(269) 764-2196

C25

Palisades Nuclear Plant Upper Housing Pressure Boundary Weld: Justification That Welds Number 3 and 4 Were Considered During CRD-24 Leakage Investigation

Introduction

During a Nuclear Regulatory Commission (NRC) special inspection of the Palisades 2012 CRD-24 Upper Housing Leak, a question was raised as to why welds number 3 and 4 were excluded from the generic implications section of the root cause evaluation. The NRC was unable to locate any technical justification as to why corrective actions for the welds were not needed. A cross-cutting finding was proposed by the NRC for not including welds 3 and 4 in the generic implications section.

The purpose of this report is to identify areas of the Root Cause Evaluation that provide evidence that welds 3 and 4 were considered during the CRD-24 Root Cause Evaluation. Additional industry guidance is referenced to justify that acceptable methods were used to prevent reoccurrence of leakage from welds 3 and 4.

ASME Section XI - IWB-2430: Requirements for Class I Components of Light-Water Cooled Plants

- 10% of peripheral CRD housings are required to be inspected every 10 years. It is permissible to defer the examination to the end of the interval. Acceptable examination methods include volumetric or surface. Palisades has 20 peripheral housings which would require 2 housings to be inspected every 10 years.
- If a flaw is discovered that exceeds the acceptance standard of Table IWB-3410-1, the inspection scope will include an additional number of inspections equal to the number of welds inspected initially. This means 2 additional housings would need to be inspected bringing the total number of housings inspected up to 4.
- If additional flaws are discovered in the expanded inspections, the remaining number of welds will be inspected. (100% of the housings would need to be inspected).
- After the inspections are complete, the examinations will return to the normally required inspections. (10% of peripheral CRD housings are required to be inspected every 10 years).
- **The inspection requirements outlined above bound welds 3 and 4 and provide adequate extent of condition. Palisades performs inspections of Class 1 Components in accordance with ASME Section XI which is approved by the NRC.**

2001 Engineering Analysis: EA-EAR-2001-0426-01

In 2001, a leak from weld 3 was identified which was caused by Stress Corrosion Cracking and manufacturing irregularities. It was decided that a replacement of all the Upper Housings was necessary. A comprehensive engineering analysis performed by Palisades and Westinghouse (EA-EAR-2001-0426-01) was completed to update the Upper Housings to prevent Stress Corrosion Cracking from welds 3 and 4 and to determine the effects of the design changes.

As part of the CRD-24 Root Cause, the design changes that were made in 2001 to prevent reoccurrence of leakage from weld 3 and 4 were discussed. These improvements included:

- *"Application of heat sink welding. The heat sink welding is a proven technology in creating a compressive residual stress on the inside surface by water-cooling while performing the welding"* and *"Enhanced surface finishing by welding shrinkage and/or honing... the final finish was required to be RMS 125 or better". (CRD-24 Root Cause, pg 7).*

- Heat sink welding is a generally accepted method (1984 EPRI Research Project T109-2) to reduce tensile stress on the interior of a weld. Making the inner diameter of a weld compressive would remove one of the elements of Transgranular Stress Corrosion Cracking and therefore would make the weld not susceptible to it.

- Engineering Analysis EA-EAR-2001-0426-01 was utilized for the Root Cause evaluation to determine that appropriate justification existed to eliminate the need for inspections of welds 3 and 4.

As part of the engineering analysis (EA-EAR-2001-0426-01) performed in 2001 to eliminate the risks of Stress Corrosion Cracking, a mockup of welds 3 and 4 were provided to Palisades for analysis by the manufacturer of the upper housings, Ionics.

Testing included:

- Visual examination, metallography, scanning electron microscopy, energy dispersive analysis, chemical analysis and hardness testing.

The metallurgical examinations (Consumers Energy, 2001, MAT Project: 0100642) performed under EA-EAR-2001-0426-01 concluded:

- *"No significant volumetric flaws were identified in the weld cross-section or adjacent base metal areas in either sample. No significant sensitization was observed". (Consumers Energy, pg 1).*

- Since no sensitization was detected in the welds, an element which was required for stress corrosion cracking, justification was provided that welds 3 and 4 did not need to be inspected.

Last Pass Heat Sink Welding Validation

An in-depth study of heat sink welding was completed by the Electric Power Research Institute (EPRI) in 1984 and concluded that:

- *"The results of this research project indicate that for pipe sizes on the order of 30.48 (12 inches) and less, LPHSW (Last Pass Heat Sink Welding) can effectively produce inside-diameter (ID) compressive residual stresses in the weld-heat-affected zone for all position welds". (EPRI 1984, pg iii).*

- This comprehensive study included destructive analysis and residual stress measurement both in the longitudinal and circumferential direction which gave Palisades additional confidence that the methods chosen to prevent leakage from reoccurring from welds 3 and 4 were valid.

2012 CRD-24 Root Cause Evaluation (CR-PLP-2012-5623)

From EN-LI-118, 5.[12].e:

- *"Perform an Extent of Cause evaluation by reviewing the individual Root and Contributing causes for generic implications".*

The root cause for the CRD-24 Upper Housing leak is:

- *"Stresses in the weld build up area due to manufacturing irregularities and misalignments between CRD-24 upper housing, support tube, seismic supports, and the associated reactor head penetration/CRDM nozzle". (CRD-24 Root Cause, pg 33).*

This uniqueness is based on the extent of condition inspections performed on 8 additional housings. No other unacceptable indications were found during the additional inspections.

The contributing cause is:

- *"Transgranular Stress Corrosion Cracking (TGSCC) initiating within the internal weld buildup material of CRD-24". (CRD-24 Root Cause, pg 33).* Because of the uniqueness of the stresses in CRD-24, TGSCC was considered for the CRD Upper Housings extent of condition and extent of cause.

The extent of condition and extent of cause were driven by the results of the Babcock and Wilcox destructive analysis of CRD-24 (PLP-RPT-12-000123). Included in the report was:

- *"Destructive examinations conducted on the nine (9) cracks identified during laboratory penetrant testing (PT) on the ID surface of the CRDM #24 housing." (CRD-24 Root Cause, pg 37).*

The pentetrant testing of the ID surface of the housing included the areas of welds 3 and 4. The upper housings were considered as a whole and therefore testing was conducted on the whole CRD-24 upper housing.

- **The dye penetrant testing of welds 3 and 4 did not show any indications of cracking even though the welds were exposed to the same conditions that promote Stress Corrosion Cracking as weld 5. This analysis drove the extent of condition of the Root Cause Evaluation and future inspection plans.**

Babcock and Wilcox concluded through destructive analysis that:

- *"Crack sizes ranged from 3" long (the thru wall crack at the "0" position) to 5/8" (lengths are approximate). All were noted to span or originate in the weld buildup area (see crack maps in the B&W report under RPT-PLP-12-000123). No circumferential cracks were identified as all identified cracks were axially located." (CRD-24 Root Cause, pg 37).*

- **Since cracking was identified only in the weld build up region in an upper housing that was known to have the conditions necessary for TGSCC to occur, it was acceptable to conclude the area around weld 5 was the only area necessary for additional inspections and welds 3 and 4 did not need inspections.**

- **It can be concluded from B&W that welds 3 and 4 were not Conditions Adverse to Quality. Any additional inspections would therefore be considered an *enhancement* to the Root Cause Analysis.**

Weld Repairs

During the CRD-24 Root Cause Analysis, all the weld repairs performed on the Upper Housings were identified and noted (CRD-24 Root Cause, pg 81-83). All of the repairs were completed in accordance with approved welding procedures as noted in the weld travelers. It was questioned whether or not the weld repairs defeated the advantages gained with last past heat sink welding.

- For all of the lower flange to pipe structure welds (weld 3), the welds were either cut out or excavated from the OD and then replaced in accordance with the approved weld procedure which would preserve the advantages of LPHSW.

Some of the upper flange to pipe structure welds (weld 4) required repair and were excavated from the ID then repaired which would increase the probability of defeating the advantages of LPHSW.

- The CRDs that were repaired in this manner include: CRD-7, 17, 21, 26, 30.

- At the time of the root cause these repairs were not considered in the generic implications but further investigation revealed that the weld repairs are not at an increased risk for Stress Corrosion Cracking.

- The water around weld 4 is at about 250 deg F and the water at weld 5 is at about 530 deg F. This makes weld 4 less susceptible to SCC than weld 5. The attached chart shows that welds that operate in a lower temperature environment are less susceptible to Stress Corrosion Cracking. This chart was obtained from a 1987 report by Dale R. Mcintyre entitled *Experience Survey Stress Corrosion Cracking of Austenitic Stainless Steels in Water*. Weld 4 requires approximately 3 times the amount of Chloride concentration that welds 3 and 5 require to promote SCC..

In the 2001 Engineering Analysis, the decision whether to use 316 or 347 SS was discussed. It was noted that:

- Using a low carbon 316 stainless steel would help to prevent stress corrosion cracking.

EPRI also published a study in 1981 that predicted the critical cooling rate that would cause sensitization during welding. The report states:

- *"....when the carbon content is reduced to less than 0.03 wt% the critical cooling rate is predicted to be less than 0.5 deg C/s. For a 0.35 in plate such a cooling rate can only be exceeded by heat inputs as large as 3937 j/mm (100,000 J/in.) A realistic heat input of 984.3 J/mm (25,000 J/in) yields a cooling rate which is 40 times larger than 0.5 deg C/s, and thus no sensitization should be (or is) noted". (EPRI 1981, pg 2-18).*

The chemical analysis of the 316SS provided in the welding travelers from 2001 for the CRD upper housings resulted in a carbon content of about 0.016 %wt.

- This is much less than 0.03 %wt which would allow much higher heat inputs to be used before sensitization occurred. **Since the welding procedure only allows a maximum heat input of 45 KJ/in, there is not an opportunity for the metal to be sensitized.**

- Therefore, even with a weld repair at weld 4, there is no appreciable sensitization of the weld and a factor of stress corrosion cracking is removed.

Discussion of Seal Housing TGSCC

- CRD Seal Housings were originally constructed of 304SS. 11 of the 48 originals cracked with 9 housings being repaired and returned to service. The repaired housings and 3 additional housings again exhibited cracking.

- It was decided to change the housings to 347SS with a post weld heat treatment to reduce residual stresses. During the fabrication of the housings, multiple weld repairs were made along with multiple post weld heat treatments. Heavy and abusive grinding was allowed after post weld heat treatment. Significant cold worked areas remained which contributed to cracking.

- Circumferential cracking in the seal housings was caused by stress from the J-weld procedure.

- Axial cracking in the seal housing was caused by residual stress from the post weld heat treatment.

- The manufacturing and welding process for the seal housings was not controlled closely as was the manufacturing of the CRD Upper Housings.

Conclusion

Based on the following information, it can be concluded that Palisades considered welds 3 and 4 during the 2012 CRD-24 Root Cause Evaluation and ASME Section XI covers the Extent of Condition;

ASME Section XI - IWB-2430: Requirements for Class I Components of Light-Water Cooled Plants

- Palisades is licensed to ASME Section XI which is approved by the NRC.

- 10% of the peripheral CRDM Upper Housing pressure boundary welds are inspected every 10 years. Volumetric inspections under Section XI bound the Extent of Condition for welds 3 and 4.

- Even if cracking and indications are found and the inspection scope is required to be expanded, the plant returns to the normal amount and interval of inspections required by Section XI following the expanded scope of inspections.

2001 Engineering Analysis of the redesigned Upper Housing:

- 2001 design changes (EA-EAR-2001-0426-01), including the improvements made to welds 3 and 4, were discussed in the Root Cause Evaluation.

- Improvements discussed included the compressive forces provided by Last Pass Heat Sink Welding and a RMS 125 surface finish which provided welds that are highly resistant to TGSCC.

- A comprehensive metallurgical analysis was also performed as a part of the engineering analysis (EA-EAR-2001-0426-01) to ensure that welds 3 and 4 would perform as required (Consumers Energy, 2001, MAT Project: 0100642).

Last Pass Heat Sink Welding:

- A 1984 EPRI study determined that Last Pass Heat Sink Welding was a valid and reliable way to produce compressive stresses on the interior of the weld. This weld process was followed in the manufacturing of welds 3 and 4.

- Testing included residual stress measurements to ensure that the welds were compressive.

2012 CRDM Root Cause Evaluation

- The generic implications section was based on the root and contributing causes for CRD-24.
- **Welds 3 and 4 were designed to prevent SCC (EA-EAR-2001-0426-01) using an industry accepted method (EPRI, 1984, Research Project T109-2).**
- Weld 5 was NOT designed to limit the sensitivity to TGSCC.
- The possibility that TGSCC could affect welds 3 and 4 was considered so the entire ID of CRD-24 was penetrant tested by Babcock and Wilcox. No indication of cracking was found.
- The results of the B&W destructive analysis drove the Extent of Condition for the Root Cause Evaluation. Welds 3 and 4 are not Adverse to Quality and additional inspections would be considered an enhancement.
- All cracking was found at weld 5, within the weld buildup area.
- No cracking was found at welds 3 and 4 in 2012 on CRD-24.
- **An environment that was conducive to TGSCC was known to exist in CRD-24. Welds 3, 4 and 5 were exposed to this environment with cracking only being found in weld 5.**
- Cracking at weld 4 was not found in the 2001 destructive analysis of the Upper Housings.
- References to the Engineer Analysis and Babcock and Wilcox were noted in the Root Cause Analysis.
- Remote visual examinations are being developed to inspect welds 3 and 4.

Weld Repairs

- As discussed in the 2001 Engineering Analysis, low carbon 316 SS was used for the Upper Housings
- Type 316SS with less than a 0.03 wt% carbon needs a very high heat input rate during welding for the material to become sensitized
- Palisades Upper Housings has 0.016 wt% which requires an extremely large amount of heat for the material to sensitize
- Weld heat inputs were limited to 45 KJ/in.

- ID weld repairs at weld 4 will not show appreciable sensitization.

CRDM Seal Housings

- 11 out of 48 original 304SS Seal Housings had circumferential cracking caused by TGSSS. Very minimal Root Cause Analysis performed.

- Weld repairs were made to the housings which caused additional stress to be applied to the housing. Heavy and abusive grinding was allowed.

- Housing material was changed to 347 with post weld heat treatment. Heavy and abusive grinding was allowed after the heat treatment which left areas of cold work.

- The post weld heat treatment left residual stresses.

- The manufacturing of the housings was poorly controlled.

In sum, the lower temperature, low carbon content, and no cracking observed in CRD-24 weld 3 and 4 provided reasonable assurance to limit the Extent of condition/extent of cause to a sample population of the weld on-lay

Additional Information

As part of Palisades review of options for inspecting the Upper Housings during the fall 2013 refueling outage, the ability to perform Eddy Current examinations of welds 3 and 4 was requested from Westinghouse in addition to eddy current testing. The Eddy Current examination will be used to examine welds 3, 4 and 5 during the inspections of the 12 CRDM housings selected for the upcoming refueling outage and will allow cracking to be identified in welds 3 and 4. Of the 12 Upper Housings included in the inspection plan for 1R23, 2 have had ID weld repairs at weld 4 which will validate the justification provided in this document.

References

2001 Engineering Analysis – EA-EAR-2001-0426-01

Babcock and Wilcox – Examinations of Cracks in CRDM #24 Housing (PLP-RPT-2012-00123)

Basic Studies on the Variabilities of Fabrication-Related Sensitization Phenomena in Stainless Steels – EPRI, 1981, Research Project 1071-1

Experience Survey Stress Corrosion Cracking of Austenitic Stainless Steels in Water, Mcintyre, Dale R. 1987

Last Pass Heat Sink Welding Process Development – EPRI, 1984, Research Project T109-2

Palisades Metallurgical Examination of CRD Weld Samples – Consumers Energy, 2001, MAT Project: 0100642

Root Cause Evaluation Report: CRD-24 Upper Housing Leak – CR-PLP-2012-05623

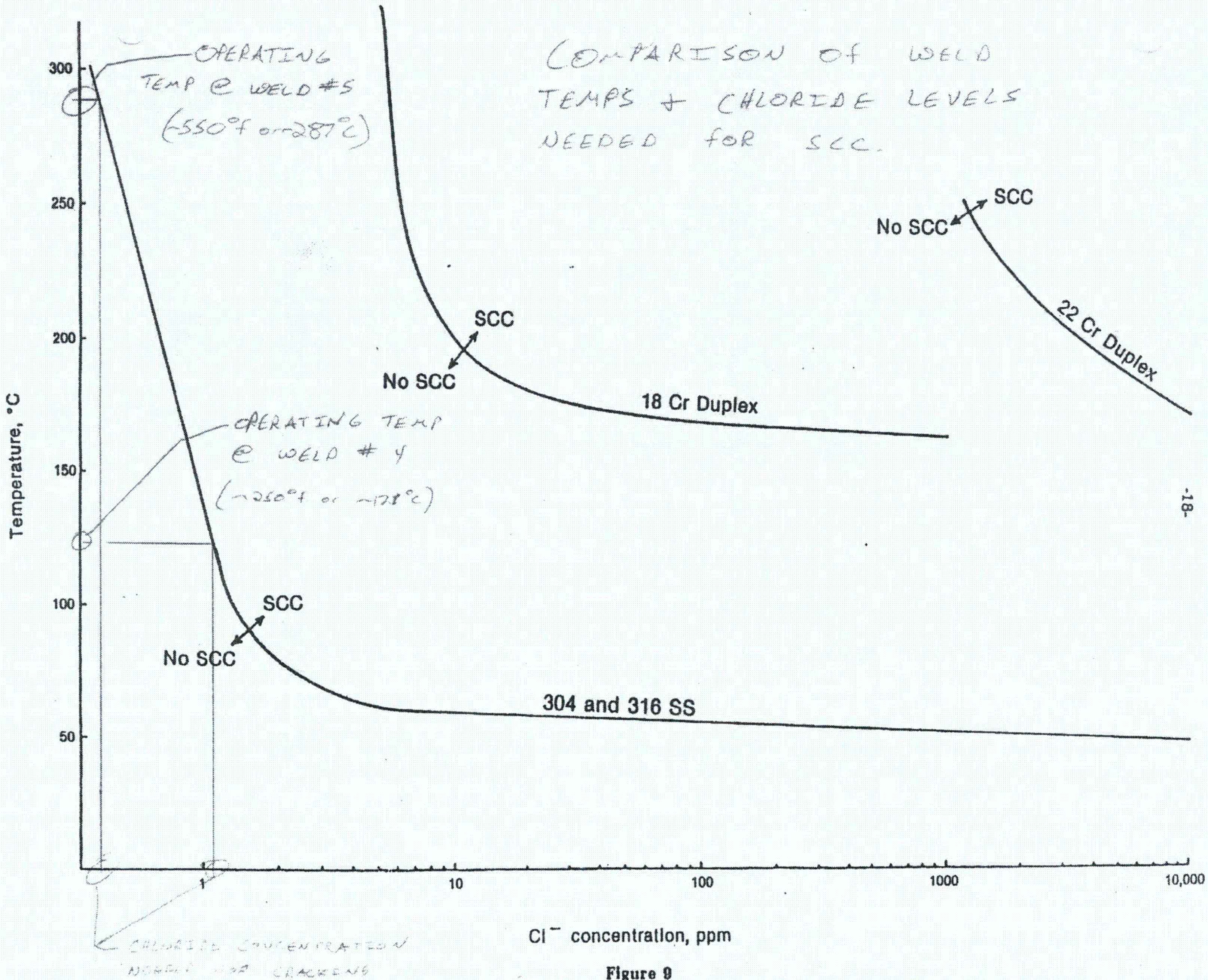

Figure 9

Relative chloride SCC resistance of solution-annealed 304 or 316 stainless steel compared to Duplex stainless steels [24].

Holmberg, Mel

From: Sanchez Santiago, Elba
Sent: Wednesday, April 17, 2013 9:19 AM
To: Alley, David; Hills, David; Holmberg, Mel; Giessner, John; Taylor, Thomas; Scarbeary, April; Betancourt, Diana
Subject: Welds 3 and 4 justification
Attachments: Welds 3 and 4 changes.docx

All,

The attached document shows the changes made to the justification based on the latest revision provided by the licensee (compared Rev 0 to Rev 1). From my initial review I didn't identify anything that would change our position. They do document that they are planning on inspecting welds 3 and 4 but consider this an enhancement and not a requirement. I will review this document again in more detail to ensure the points they are making do not invalidate our assumptions and alleviate our concerns. Let me know if you have any comments or questions.

Thanks,
Elba

Note: We are planning on formally exiting this issue tomorrow with a NCV of Criterion V for failure to follow the root cause procedure. Specifically, the failure to adequately evaluate the generic implications of the cause of the cracking identified in CRDM No. 24.

C26

Palisades Nuclear Plant Upper Housing Pressure Boundary Weld: Justification That Welds Number 3 and 4 Were Considered During CRD-24 Leakage Investigation

Introduction

During a Nuclear Regulatory Commission (NRC) special inspection of the Palisades 2012 CRD-24 Upper Housing Leak, a question was raised as to why welds number 3 and 4 were excluded from the generic implications section of the root cause evaluation. The NRC was unable to locate any technical justification as to why corrective actions for the welds were not needed. A cross-cutting finding was proposed by the NRC for not including welds 3 and 4 in the generic implications section.

The purpose of this report is to identify areas of the Root Cause Evaluation that provide evidence that welds 3 and 4 were considered during the CRD-24 Root Cause Evaluation. Additional industry guidance is referenced to justify that acceptable methods were used to prevent reoccurrence of leakage from welds 3 and 4.

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ASME Section XI - IWB-2430: Requirements for Class I Components of Light-Water Cooled Plants

- 10% of peripheral CRD housings are required to be inspected every 10 years. It is permissible to defer the examination to the end of the interval. Acceptable examination methods include volumetric or surface. Palisades has 20 peripheral housings which would require 2 housings to be inspected every 10 years.

- If a flaw is discovered that exceeds the acceptance standard of Table IWB-3410-1, the inspection scope will include an additional number of inspections equal to the number of welds inspected initially. This means 2 additional housings would need to be inspected bringing the total number of housings inspected up to 4.

- If additional flaws are discovered in the expanded inspections, the remaining number of welds will be inspected. (100% of the housings would need to be inspected).

- After the inspections are complete, the examinations will return to the normally required inspections. (10% of peripheral CRD housings are required to be inspected every 10 years).

- **The inspection requirements outlined above bound welds 3 and 4 and provide adequate extent of condition. Palisades performs inspections of Class 1 Components in accordance with ASME Section XI which is approved by the NRC.**

2001 Engineering Analysis: EA-EAR-2001-0426-01

In 2001, a leak from weld 3 was identified which was caused by Stress Corrosion Cracking and manufacturing irregularities. It was decided that a replacement of all the Upper Housings was necessary. A comprehensive engineering analysis performed by Palisades and Westinghouse (EA-EAR-2001-0426-01) was completed to update the Upper Housings to prevent Stress Corrosion Cracking from welds 3 and 4 and to determine the effects of the design changes.

As part of the CRD-24 Root Cause, the design changes that were made in 2001 to prevent reoccurrence of leakage from weld 3 and 4 were discussed. These improvements included:

- *"Application of heat sink welding. The heat sink welding is a proven technology in creating a compressive residual stress on the inside surface by water-cooling while performing the welding"* and *"Enhanced surface finishing by welding shrinkage and/or honing... the final finish was required to be RMS 125 or better". (CRD-24 Root Cause, pg 7).*

- Heat sink welding is a generally accepted method (1984 EPRI Research Project ~~1071-1~~T109-2) to reduce tensile stress on the interior of a weld. Making the inner diameter of a weld compressive would remove one of the elements of Transgranular Stress Corrosion Cracking and therefore would make the weld not susceptible to it.

- Engineering Analysis EA-EAR-2001-0426-01 was utilized for the Root Cause evaluation to determine that appropriate justification existed to eliminate the need for inspections of welds 3 and 4.

As part of the engineering analysis (EA-EAR-2001-0426-01) performed in 2001 to eliminate the risks of Stress Corrosion Cracking, a mockup of welds 3 and 4 were provided to Palisades for analysis by the manufacturer of the upper housings, Ionics.

Testing included:

- Visual examination, metallography, scanning electron microscopy, energy dispersive analysis, chemical analysis and hardness testing.

The metallurgical examinations (Consumers Energy, 2001, MAT Project: 0100642) performed under EA-EAR-2001-0426-01 concluded:

- *"No significant volumetric flaws were identified in the weld cross-section or adjacent base metal areas in either sample. No significant sensitization was observed". (Consumers Energy, pg 1).*

- Since no sensitization was detected in the welds, an element which was required for stress corrosion cracking, justification was provided that welds 3 and 4 did not need to be inspected.

Last Pass Heat Sink Welding Validation

An in-depth study of heat sink welding was completed by the Electric Power Research Institute (EPRI) in 1984 and concluded that:

- *"The results of this research project indicate that for pipe sizes on the order of 30.48 (12 inches) and less, LPHSW (Last Pass Heat Sink Welding) can effectively produce inside-diameter (ID) compressive residual stresses in the weld-heat-affected zone for all position welds". (EPRI 1984, pg iii).*

- This comprehensive study included destructive analysis and residual stress measurement both in the longitudinal and circumferential direction which gave Palisades additional confidence that the methods chosen to prevent leakage from reoccurring from welds 3 and 4 were valid.

2012 CRD-24 Root Cause Evaluation (CR-PLP-2012-5623)

From EN-LI-118, 5.[12].e:

- *"Perform an Extent of Cause evaluation by reviewing the individual Root and Contributing causes for generic implications".*

The root cause for the CRD-24 Upper Housing leak is:

- *"Stresses in the weld build up area due to manufacturing irregularities and misalignments between CRD-24 upper housing, support tube, seismic supports, and the associated reactor head penetration/CRDM nozzle". (CRD-24 Root Cause, pg 33).*

This uniqueness is based on the extent of condition inspections performed on 8 additional housings. No other unacceptable indications were found during the additional inspections.

The contributing cause is:

- *"Transgranular Stress Corrosion Cracking (TGSCC) initiating within the internal weld buildup material of CRD-24". (CRD-24 Root Cause, pg 33).* Because of the uniqueness of the stresses in CRD-24, TGSCC was considered for the CRD Upper Housings extent of condition and extent of cause.

The extent of condition and extent of cause were ~~based on~~driven by the results of the Babcock and Wilcox destructive analysis of CRD-24 (PLP-RPT-12-000123). Included in the report was:

- *"Destructive examinations conducted on the nine (9) cracks identified during laboratory penetrant testing (PT) on the ID surface of the CRDM #24 housing." (CRD-24 Root Cause, pg 37).*

The pentetrant testing of the ID surface of the housing included the areas of welds 3 and 4. The upper housings were considered as a whole and therefore testing was conducted on the whole CRD-24 upper housing.

- **The dye penetrant testing of welds 3 and 4 did not show any indications of cracking even though the welds were exposed to the same conditions that promote Stress Corrosion Cracking as weld 5. This analysis drove the extent of condition of the Root Cause Evaluation and future inspection plans.**

Babcock and Wilcox concluded through destructive analysis that:

- *"Crack sizes ranged from 3" long (the thru wall crack at the "0" position) to 5/8" (lengths are approximate). All were noted to span or originate in the weld buildup area (see crack maps in the B&W report under RPT-PLP-12-000123). No circumferential cracks were identified as all identified cracks were axially located." (CRD-24 Root Cause, pg 37).*

- **Since cracking was identified only in the weld build up region in an upper housing that was known to have the conditions necessary for TGSCC to occur, it was acceptable to conclude the area around weld 5 was the only area necessary for additional inspections and welds 3 and 4 did not need inspections.**

- **It can be concluded from B&W that welds 3 and 4 were not Conditions Adverse to Quality. Any additional inspections would therefore be considered an *enhancement* to the Root Cause Analysis.**

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Weld Repairs

During the CRD-24 Root Cause Analysis, all the weld repairs performed on the Upper Housings were identified and noted (CRD-24 Root Cause, pg 81-83). All of the repairs were completed in accordance with approved welding procedures as noted in the weld travelers. It was questioned whether or not the weld repairs defeated the advantages gained with last past heat sink welding.

- For all of the lower flange to pipe structure welds (weld 3), the welds were either cut out or excavated from the OD and then replaced in accordance with the approved weld procedure which would preserve the advantages of LPHSW.

Some of the upper flange to pipe structure welds (weld 4) required repair and were excavated from the ID then repaired which would increase the probability of defeating the advantages of LPHSW.

- The CRDs that were repaired in this manner include: CRD-7, ~~12,~~ 17, 21, 26, ~~29,~~ 30~~, 32, and 42~~.

- At the time of the root cause these repairs were not considered in the generic implications but further investigation revealed that the weld repairs are not at an increased risk for Stress Corrosion Cracking.

- The water around weld 4 is at about 250 deg F and the water at weld 5 is at about 530 deg F. This makes weld 4 less susceptible to SCC than weld 5. The attached chart shows that welds that operate in a lower temperature environment are less susceptible to Stress Corrosion Cracking. This chart was obtained from a 1987 report by Dale R. Mcintyre entitled *Experience Survey Stress Corrosion Cracking of Austenitic Stainless Steels in Water*. Weld 4 requires approximately 3 times the amount of Chloride concentration that welds 3 and 5 require to promote SCC.

In the 2001 Engineering Analysis, the decision whether to use 316 or 347 SS was discussed. It was noted that:

- Using a low carbon 316 stainless steel would help to prevent stress corrosion cracking.

EPRI also published a study in 1981 that predicted the critical cooling rate that would cause sensitization during welding. The report states:

- *"....when the carbon content is reduced to less than 0.03 wt% the critical cooling rate is predicted to be less than 0.5 deg C/s. For a 0.35 in plate such a cooling rate can only be exceeded by heat inputs as large as 3937 j/mm (100,000 J/in.) A realistic heat input of 984.3 J/mm (25,000 J/in) yields a cooling rate which is 40 times larger than 0.5 deg C/s, and thus no sensitization should be (or is) noted". (EPRI 1981, pg 2-18).*

The chemical analysis of the 316SS provided in the welding travelers from 2001 for the CRD upper housings resulted in a carbon content of about 0.016 %wt.

- This is much less than 0.03 %wt which would allow much higher heat inputs to be used before sensitization occurred. **Since the welding procedure only allows a maximum heat input of 45 KJ/in, there is not an opportunity for the metal to be sensitized.**

- Therefore, even with a weld repair at weld 4, there is no appreciable sensitization of the weld and a factor of stress corrosion cracking is removed.

Discussion of Seal Housing TGSCC

- CRD Seal Housings were originally constructed of 304SS. 11 of the 48 originals cracked with 9 housings being repaired and returned to service. The repaired housings and 3 additional housings again exhibited cracking.

- It was decided to change the housings to 347SS with a post weld heat treatment to reduce residual stresses. During the fabrication of the housings, multiple weld repairs were made along with multiple post weld heat treatments. Heavy and abusive grinding was allowed after post weld heat treatment. Significant cold worked areas remained which contributed to cracking.

- Circumferential cracking in the seal housings was caused by stress from the J-weld procedure.

- Axial cracking in the seal housing was caused by residual stress from the post weld heat treatment.

- The manufacturing and welding process for the seal housings was not controlled closely as was the manufacturing of the CRD Upper Housings.

Conclusion

Based on the following information, it can be concluded that Palisades considered welds 3 and 4 during the 2012 CRD-24 Root Cause Evaluation and ASME Section XI covers the Extent of Condition;

ASME Section XI - IWB-2430: Requirements for Class 1 Components of Light-Water Cooled Plants

- Palisades is licensed to ASME Section XI which is approved by the NRC.

- 10% of the peripheral CRDM Upper Housing pressure boundary welds are inspected every 10 years. Volumetric inspections under Section XI bound the Extent of Condition for welds 3 and 4.

- Even if cracking and indications are found and the inspection scope is required to be expanded, the plant returns to the normal amount and interval of inspections required by Section XI following the expanded scope of inspections.

2001 Engineering Analysis of the redesigned Upper Housing:

- 2001 design changes (EA-EAR-2001-0426-01), including the improvements made to welds 3 and 4, were discussed in the Root Cause Evaluation.

- Improvements discussed included the compressive forces provided by Last Pass Heat Sink Welding and a RMS 125 surface finish which provided welds that are highly resistant to TGSCC.

- A comprehensive metallurgical analysis was also performed as a part of the engineering analysis (EA-EAR-2001-0426-01) to ensure that welds 3 and 4 would perform as required (Consumers Energy, 2001, MAT Project: 0100642).

Last Pass Heat Sink Welding:

- A 1984 EPRI study determined that Last Pass Heat Sink Welding was a valid and reliable way to produce compressive stresses on the interior of the weld. This weld process was followed in the manufacturing of welds 3 and 4.

- Testing included residual stress measurements to ensure that the welds were compressive.

2012 CRDM Root Cause Evaluation

- The generic implications section was based on the root and contributing causes for CRD-24.

- **Welds 3 and 4 were designed to prevent SCC (EA-EAR-2001-0426-01) using an industry accepted method (EPRI, ~~1987~~1984, Research Project T109-2).**

- Weld 5 was NOT designed to limit the sensitivity to TGSCC.

- The possibility that TGSCC could affect welds 3 and 4 was considered so the entire ID of CRD-24 was penetrant tested by Babcock and Wilcox. No indication of cracking was found.

- The results of the B&W destructive analysis drove the Extent of Condition for the Root Cause Evaluation. Welds 3 and 4 are not Adverse to Quality and additional inspections would be considered an enhancement.

- All cracking was found at weld 5, within the weld buildup area.

- No cracking was found at welds 3 and 4 in 2012 on CRD-24.

- **An environment that was conducive to TGSCC was known to exist in CRD-24. Welds 3, 4 and 5 were exposed to this environment with cracking only being found in weld 5.**

- Cracking at weld 4 was not found in the 2001 destructive analysis of the Upper Housings.

- References to the Engineer Analysis and Babcock and Wilcox were noted in the Root Cause Analysis.

- Remote visual examinations are being developed to inspect welds 3 and 4.

Weld Repairs

- As discussed in the 2001 Engineering Analysis, low carbon 316 SS was used for the Upper Housings

- Type 316SS with less than a 0.03 wt% carbon needs a very high heat input rate during welding for the material to become sensitized

- Palisades Upper Housings has 0.016 wt% which requires an extremely large amount of heat for the material to sensitize

- Weld heat inputs were limited to 45 KJ/in.

- ID weld repairs at weld 4 ~~are not sensitized~~will not show appreciable sensitization.

CRDM Seal Housings

- 11 out of 48 original 304SS Seal Housings had circumferential cracking caused by TGSSS. Very minimal Root Cause Analysis performed.

- Weld repairs were made to the housings which caused additional stress to be applied to the housing. Heavy and abusive grinding was allowed.

- Housing material was changed to 347 with post weld heat treatment. Heavy and abusive grinding was allowed after the heat treatment which left areas of cold work.

- The post weld heat treatment left residual stresses.

- The manufacturing of the housings was poorly controlled.

In sum, the lower temperature, low carbon content, and no cracking observed in CRD-24 weld 3 and 4 provided reasonable assurance to limit the Extent of condition/extent of cause to a sample population of the weld on-lay

Additional Information

As part of Palisades review of options for inspecting the Upper Housings during the fall 2013 refueling outage, the ability to perform ~~remote visual~~Eddy Current examinations of welds 3 and 4 was requested ~~in January of 2013~~ from Westinghouse in addition to eddy current testing. The ~~visual~~Eddy Current examination will be used to examine welds 3, 4 and 5 during the inspections of the 12 CRDM housings selected for the upcoming refueling outage and will allow cracking to be identified in welds 3 and 4. Of the 12 Upper Housings included in the inspection plan for 1R23, ~~3~~2 have had ID weld repairs at weld 4 which will validate the justification provided in this document.

~~Babcock and Wilcox took extensive photos of the CRD-24 Upper Housing during destructive analysis including the cracks in weld 5 before penetrant testing. The cracks at weld 5 in the CRD-24 Upper Housing were able to be distinguished in as-found photos taken with a digital camera. Therefore, Palisades has high confidence that if cracking in welds 3 and 4 has occurred then they can be easily found through visual exams.~~

References

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2001 Engineering Analysis – EA-EAR-2001-0426-01

Babcock and Wilcox – Examinations of Cracks in CRDM #24 Housing (PLP-RPT-2012-00123)

Basic Studies on the Variabilities of Fabrication-Related Sensitization Phenomena in Stainless Steels – EPRI, 1981, Research Project 1071-1

Experience Survey Stress Corrosion Cracking of Austenitic Stainless Steels in Water, Mcintyre, Dale R. 1987

Last Pass Heat Sink Welding Process Development – EPRI, ~~1987~~1984, Research Project T109-2

Palisades Metallurgical Examination of CRD Weld Samples – Consumers Energy, 2001, MAT Project: 0100642

Root Cause Evaluation Report: CRD-24 Upper Housing Leak – CR-PLP-2012-05623

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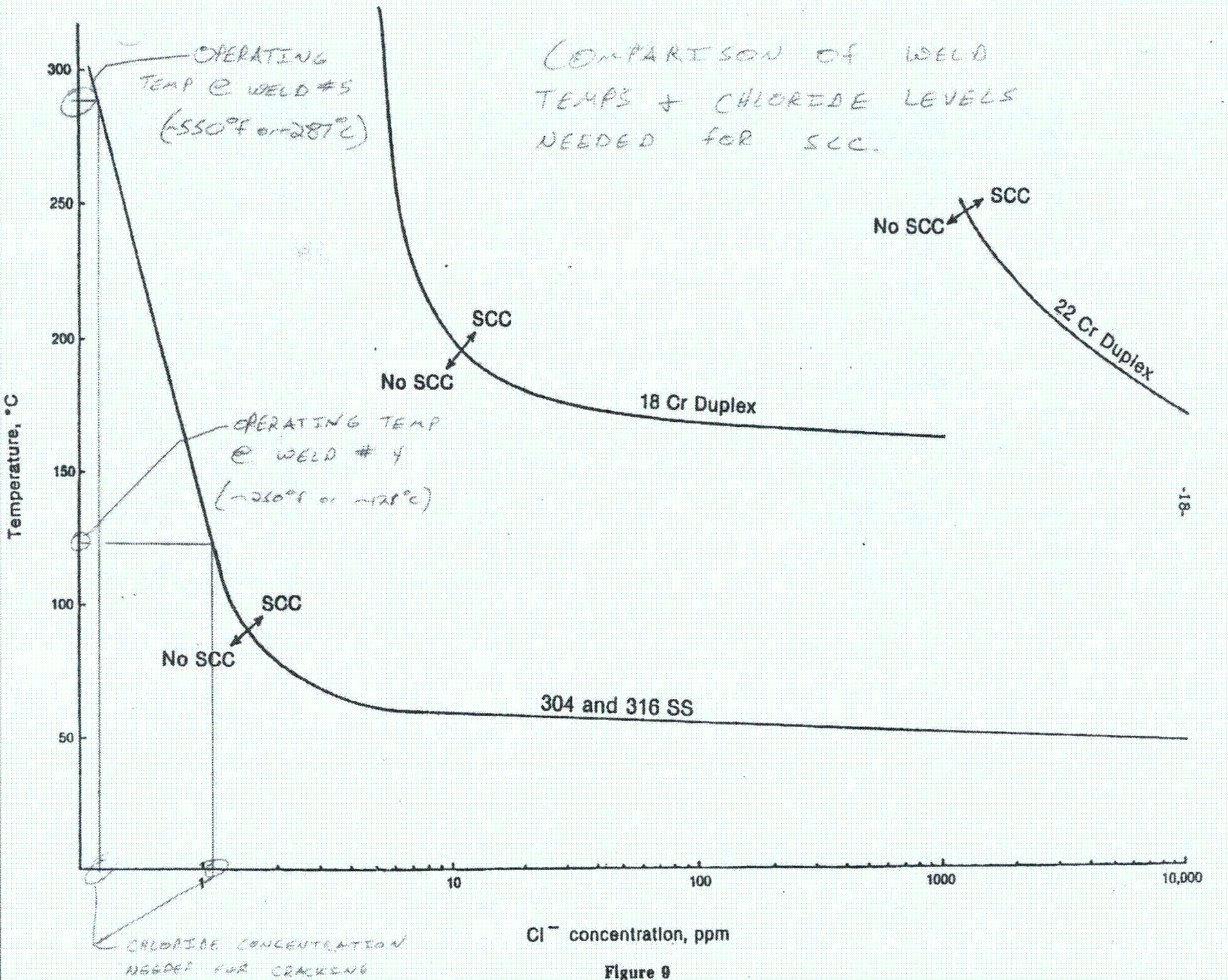

Figure 9
Relative chloride SCC resistance of solution-annealed 304 or 316 stainless steel compared to Duplex stainless steels [24].

Giessner, John RELEASE ENTIRETY

From: Sanchez Santiago, Elba
Sent: Thursday, April 18, 2013 3:53 PM
To: Giessner, John
Subject: Palisades exit notes
Attachments: palisades final exit.docx

Jack,

Attached are the palisades exit notes I used during today's exit meeting. Let me know if you have any questions.

Just FYI I confirmed Barry Davis was not at the exit. Jody Haumerson was there as the acting engineering director.

-Elba

C17

1. URI Crack growth rate:

   Dave Alley and I performed a follow-up inspection to determine if the assumptions you made were conservative and the planned actions bounded those conservative assumptions. We reviewed a variety of documents associated with crack growth and inspection intervals and noted the following statements included in the root cause report and vendor documents related to the determination of the appropriate crack growth rate.

   1. The lab conducting the failure analysis concluded, it could not be conclusively determined if the beach marks corresponded to refueling outages, (i.e., 18 month cycle) or shorter periods as occurred during outages over the past 24 months
   2. Palisades CRDM housing 21 leaked at weld 3 in 2001. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 failure. In calculating the crack growth rate of this crack, one contractor utilized an interval between beach marks which is much shorter than refueling outages. The intervals used are consistent with plant thermal cycles in which oxygen may or may not have been admitted into the CRDMs.
   3. A spare CRDM housing at Ft Calhoun leaked at weld 5 in 1990. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 failure. In calculating the crack growth rate of this crack, Ft Calhoun stated that the beach marks were related to refueling cycles. Ft Calhoun also performed calculations indicating that the oxygen level at the location of the flaw did not change with time (including in response to refueling outages) because the spare CRDM housing was not vented. Ft Calhoun's evaluation indicated that oxygen levels at the vicinity of the crack would have begun to decline through diffusion and convection had the intervals between outages been much longer than 18 months. This is interpreted to mean that the beach marks at Ft Calhoun are in response to pressure/thermal cycles.
   4. In at least one instance Palisades needed to repair the seals on a reactor coolant pump at a time other than an outage. This necessitates draining some of the water from the reactor coolant system and venting (admitting oxygen into) the CRDM housing. This represents an additional oxygen ingress event not included when determination of time to cracking is based on refueling outages.
   5. In its inspection plan Palisades states that it will inspect all CRDM housings over the next 4 refueling outages, i.e., the interval between inspections is 1 refueling outage

   Based on the above review, we noted that there are certain non conservative statements contained in the Root Cause Report and the inspection plan. These include:

   1. The crack growth rate based on refueling outages is understated. If oxygen ingress is related to beach marks, given the oxygen ingress event which occurred to repair reactor coolant pump seals, 6 beach marks would occur in a maximum of 5 refueling intervals rather than the 6 refueling intervals that were used to calculate the crack growth rate in the root cause report.
   2. The crack growth rate based on heat up and cool down cycles is overstated. The value in the root cause is based on 11 months. While 6 shutdowns did occur at the plant in 11 months several of these events did not result in pressure/temperature changes of the reactor coolant system. The appropriate time frame is 24 months rather than 11.

3. The inspection plan contains a non conservative statement: "However, once the crack has been initiated it propagates over 4 to 5 operating cycles prior to going through wall." While this statement does reflect one of the proposed theories for crack growth, sufficient evidence to demonstrate reasonable assurance that this theory is correct, and thereby overcome the non-conservatism of this statement, does not exist.

Despite the existence of the non conservatisms stated above, we concluded:

1. Sufficient evidence to conclusively determine the rate of crack growth does, and will not exist.
2. Crack growth based on pressure/temperature cycles is the most conservative of the potential crack growth mechanisms. In the absence of reasonable assurance of the correctness of less conservative mechanisms, through wall crack growth in 2 years must be utilized.
3. The licensee has not formally committed to any of the crack growth mechanisms discussed.
4. Your inspection program includes inspections in each of the next 4 outages. This inspection interval, once per outage, bounds all the crack growth mechanisms considered.

We find this approach to inspection to be both acceptable and sufficient justification to close this URI.

2. Unresolved Item Failure to prevent recurrence and technical specifications violation for operating with pressure boundary leakage.
    a. When the leak was identified in 2001 various corrective actions were applied to prevent recurrence. These corrective actions were limited to pressure boundary welds and the need for corrective actions related to weld 5 was not considered.
    b. Based on our review of your 2001 root cause report, 2012 root cause report, various vendor documents and interviews with your staff we identified you failed to eliminate one or more of the necessary factors at the weld build-up area to preclude TGSCC in the replacement housings. Specifically
        i. The 2001 root cause report documented weld 5 is exposed essentially to the same environment as weld 3.
        ii. Fabrication restrictions to prohibit grinding were not applied to the weld build-up region which promoted residual tensile stresses on the ID of the CRDM surface.
        iii. Material was changed from type 347 to type 316 stainless steel which are essentially equal as far as resistance to TGSCC
    c. Based on the recurrence of through wall leakage in the CRDM housings that occurred at the weld build-up region of the CRDM housings by TGSCC we concluded that the actions taken in 2001 were not adequate because the appropriate actions to preclude recurrence were within your ability to foresee and implement.
    d. We identified a performance deficiency for failure to prevent recurrence of a significant condition adverse to quality resulting in a non-compliance with the TS.

e. More than minor because it adversely affects the initiating events cornerstone objective for not limiting the likelihood of events that upset plant stability, specifically the cornerstone attribute of equipment performance.
f. Because this issue was entered into your corrective action program we identified this as a NCV of 10 CFR appendix B Criterion XVI "corrective actions" and Technical specification 3.4.13 "primary Coolant System Operational Leakage" for failure to prevent recurrence of leakage in CRDM housings due to TGSCC resulting in the operation of the reactor with pressure boundary leakage for greater than the TS allowed time.
g. The performance deficiency screened as green after screening under the initiating events cornerstone because we answered no to the question if after reasonable assessment of degradation, could the finding result in exceeding the RCS leak rate for a small LOCA and could the finding have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. Basically because of the nature of the cracking and your site procedures we believe it would be detected and corrected prior to reaching the small break LOCA limits.
h. We are not recommending a cross cutting aspect be applied to this performance deficiency though we do believe this issue is still indicative of current performance we are addressing this aspect in the next violation we will discuss.

3. Violation of root cause procedure
   a. When reviewing the 2012 root cause procedure related to the cracking identified in CRDM 24 we identified a failure to appropriately consider the generic implications of the cracking in the extent of condition review. The proposed corrective actions narrowly focused on the weld build up region instead of broader actions to ensure other CRDM housing welds were fit for their intended service life.
   b. You provided additional information to us to justify excluding these welds from the scope of the corrective actions.
   c. You credited the actions taken in 2001 and stated that these actions would produce compressive stresses on the ID of welds 3 and 4 making them immune to cracking. These actions included performing heat sink welding,, changing th design around weld 3 and specifying a smoother surface finish.
   d. We acknowledge that these actions would reduce the tensile stress at the ID surface and thus reduce the probability of initiating TGSCC
   e. However the information provided did not demonstrate that TGSCC would not occur because it did not demonstrate that tensile stress would be eliminated at the ID surface during operation. In particular repairs completed at the inner surface of weld 4 would result in high residual tensile stress at the inside surface of the weld which would promote the initiation of TGSCC.
   f. Repairs were also performed on weld 3 from the outer diameter surface of the weld and the assumption has been made that heat sink welding would be sufficient to ensure residual compressive stress would remain at the ID surface of weld 3 even

with repairs to the OD surface. However a detailed residual stress test or modeling has not been performed to confirm this assumption.

g. We identified that the three factors required for TGSCC could still be present at welds 3 and 4

- Corrosive environment – Weld 3 would operate in a similar environment as the weld build up region of the CRDM housing. Weld No. 4 would be exposed to a lower operating temperature then the weld build up region, however TGSCC can still occur at 250 degrees Fahrenheit as evidenced by the previous operating experience with cracking identified in the seal housings that operate at even lower temperatures.

- Susceptible material – Welds 3, 4 and 5 are composed of the same weld filler and base metal materials as the weld buildup region (e.g. weld filler material consistent with the type 316 stainless housing base metal). This material would be equally susceptible to TGSCC, as the type 347 stainless steel and weld filler materials used in the pre-2001 CRDM housing design that developed a through wall leak caused by TGSCC at weld No.3.

- Tensile stresses - While it is assumed that the corrective actions taken in response to the 2001 leak will reduce the potential for tensile stresses to exist on the inner surface of CRDM housings at welds 3 and 4, especially in light of the repairs made to welds 3 and 4, it has not been conclusively demonstrated that these tensile stresses have been eliminated. As such it is not reasonable to conclude that tensile stresses are not present and, therefore, the potential for transgranular stress corrosion cracking has been eliminated.

h. The discussion of sensitization is not germane to the observed cracking. Sensitization is not required for transgranular stress corrosion cracking. The use of materials resistant to sensitization do not reduce the likelihood of transgranular stress corrosion cracking.

i. Despite the fact that the root cause for the leak in CRDM housing 24 indicts manufacturing issues and alignment, it also includes an unidentified stress. This stress, if it exists, may be present to a greater or lesser extent in other housings. Based on this, it is not clear that the absence of cracking in welds 3 and 4 of CRDM housing 24 is definitive evidence that welds 3 and 4 are not subject to cracking in other CRDM housings. It also should be noted that CRDM housing 24 is not listed as having undergone weld repairs to weld 4. Cracking at weld 4 is more likely in CRDM housings other than CRDM 24.Therefore we do not believe you have established a sufficient basis in the RCR to exclude welds 3 and 4 from the extent of condition review

j. We determined that the failure to adequately evaluate and document the generic implications of the cause of cracking identified in CRDM #23 in accordance with the root cause procedure EN-LI-118 was a performance deficiency.

k. We determined this issue was more than minor because it adversely affected the Initiating event cornerstone attribute of equipment performance and we answered yes to the question if left uncorrected would the performance deficiency have the potential to lead to a more significant safety concern.

l. Specifically, absent NRC identification, you would not have completed further evaluations or inspections of CRDM housing welds which could have resulted in additional CRDM housing failure and leakage by TGSCC

m. The performance deficiency screened as green after screening under the initiating events cornerstone because we answered no to the question if after reasonable assessment of degradation, could the finding result in exceeding the RCS leak rate for a small LOCA and could the finding have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. Basically because of the nature of the cracking and your site procedures we believe it would be detected and corrected prior to reaching the small break LOCA limits.

n. We are recommending a Cross cutting aspect in the area of human performance Decision Making, because conservative assumptions were not used in decision making. Specifically, did not use conservative assumptions when excluding welds 3 and 4 as being susceptible to TGSCC and therefore include them in the generic implications section of the root cause report. (Item H.1(b))

o. Because you entered this issue into your corrective action program we are characterizing this issue as a NCV of 10 CFR Appendix B Criterion V "Instructions, Procedures and Drawings", having a very low safety significance (Green), for failure to adequately evaluate the generic implications of the cause of cracking identified in CRDM #24 as it relates to weld 3 and 4 in accordance with the root cause procedure.

p. Title 10 CFR Appendix B Criterion V "Instruction, Procedures and Drawings requires in part, "Activities affecting quality shall be prescribed by documented procedures and shall be accomplished in accordance with these procedures."

q. Procedure EN-LI-118 Root Cause evaluation process revision 17 states:

    a. 5.5 (12)e: perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affects other SSC's , organizations or work processes. Use the two step process in accordance with attachment 9.7

    b. Attachment 9.7 states Determine whether the occurrence/consequence (problem) is isolated, or whether it has broader (generic or common mode) implications. Achieve this by asking the following questions:

        i. Could this happen to equipment that is similar in function, design, or service condition?

        ii. Could this happen to a group of components? (components of the same construction or materials that could be similarly affected by one condition)

    c. Attachment 9.7 also states: Document the results of the above considerations. Include the following items in the write up:

i. Generic Implications (Is this problem/ cause limited to this component/equipment, or does it apply to others as well)

ii. Existing broader (generic/common mode) considerations

d. 5.5(15)(10)c&f: Document proposed corrective actions and due dates to address valid generic implications. If no corrective action is recommended for a valid generic implication then document the basis for this conclusion and any risk or consequence identified as a result of taking no action.

r. Contrary to the above, you failed to perform an activity affecting quality in accordance with procedure EN-LI-118. Specifically, you did not identify and document the existing broader (generic/common mode) considerations associated with TGSCC at CRDM housing welds No. 3 and No. 4. Consequently, you failed to propose corrective actions for the generic implications of TGSCC at CRDM housing welds No. 3 and No. 4.

Questions?

In order for us to follow our process I wanted to take this opportunity to ask if you are planning on contesting any of these violations.

Thanks You

Hills, David

From: Sanchez Santiago, Elba
Sent: Thursday, April 18, 2013 8:20 AM
To: Hills, David
Subject: Palisades write-up
Attachments: Palisades Input to DRP Report 2013 002 URI EMS.docx

Dave,

Attached is the palisades input for your review. I haven't attached the documents reviewed section yet but wanted to send this out so you could review the write-up and characterization of the findings and the URI we are closing out. Let me know if you have any questions or comments.

Thanks,
Elba

C-8

UNITED STATES
NUCLEAR REGULATORY COMMISSION
LISLE, IL 60532-4352

April XX, 2012

MEMORANDUM TO: Thomas Taylor
Senior Resident Inspector
Palisades Nuclear Plant

FROM: David Hills, Chief
Engineering Branch 3
Division of Reactor Safety

SUBJECT: PALISADES NUCLEAR PLANT
DRS INPUT TO INTEGRATED REPORT 05000255/2013002

Enclosed is the report input for the Palisades Nuclear Plant, Inspection Report 05000255/2013002. This report input documents completion of our review of Unresolved Items 05000255/2012012-01, "TS for PCS Pressure Boundary Leakage," 05000255/2012012-02, "Potential Inadequate Degradation Evaluation of CRDM Housings," and 05000255/2012012-03, "Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality." This report also completes the one sample of the Problem Identification and Resolution, Selected Issue Follow-up in accordance with IP 71152. I have reviewed this input to confirm compliance with Inspection Manual Chapter (IMC) 0612 and IMC 305. This input is ready for inclusion into the integrated report and dissemination to the public.

Please input the following post Inspection Data into RPS:

Inspection Procedure	**Procedure Status** – see below: Incomplete, Complete, Complete by reference, Complete-full sample not available, Complete – opportunity to apply procedure not available, Not Applicable.	**Sample Size** – As documented in Scope Section If less than full sample size documented in the report input, the inspector must provide a justification below to enter into RPS and support the procedure status selected
71152	Complete	1

Inspection Report Item and Type (AV, FIN, NCV, URI or VIO)	**Cornerstone** (IE, MS, BI, EP, OR, PR, MISC)	**Cross Cutting Aspect** (H.n(i), P.n(i), S.n(i))	**Responsible Person/Owner**	**Procedure or TI** (71111.07T)	**RPS Branch Code (e.g. closeout responsibility)** EB1 3820 EB2 3870 EB3 3840 PST (RP) 3860 PSB (Safeguards) 3850 OB 3810
NCV-XXX	IE	n/a	E. Sanchez Santiago	71152	3820
NCV-XXX	IE	H.1(b)	E. Sanchez Santiago	71152	3820

Enclosure: Input to Inspection Report 05000255/2013002

cc w/encl: J. Giessner, Chief
C. Hernandez, Site Admin Assistant

CONTACT: E. Sanchez Santiago, DRS
(630) 829-9715

DOCUMENT NAME: G:\DRSIII\DRS\Work in Progress\-Palisades Input to DRP Report 2013 002 URI EMS.docx
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OFFICE	RIII		NRR		RIII		NRR	
NAME	ESanchezSantiago		DAlley		DHills		TLupold	
DATE	4/ /13							

OFFICIAL RECORD COPY

Cover Letter

X Green findings involving a violation were identified. Include the following:

Based on the results of this inspection, two NRC-identified findings of very low safety significance (Green) were identified. These findings were determined to involve a violation of NRC requirements. However, because of the very low safety significance and because the issues were entered into your corrective action program, the NRC is treating the issue as Non-Cited Violation, in accordance with Section 2.3.2 of the NRC Enforcement Policy.

TITLE PAGE

Inspectors: D. Alley, Senior Materials Engineer
E. Sanchez Santiago, Reactor Inspector

SUMMARY OF FINDINGS

A. NRC-Identified and Self-Revealed Findings

Cornerstones: Initiating Events

- Green. The inspectors identified a Green Non-Cited Violation (NCV) of 10 CFR Part 50 Appendix B Criterion XVI, and Technical Specification (TS) 3.4.14 Primary Coolant System (PCS) Operational Leakage, for failure to prevent recurrence of a significant condition adverse to quality resulting in a non-compliance with the TS. Specifically, the licensee failed to include the internal CRDM weld build-up area within the scope of corrective actions taken for a 2001 CRDM housing leakage event (a significant condition adverse to quality) and consequently leakage recurred at the CRDM housing No. 24 in 2012.

  The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. Specifically the licensee did not limit the likelihood of events that upset plant stability by not taking adequate corrective actions to prevent recurrence of leakage in CRDM housings which represents a pressure boundary leakage and a condition prohibited by the technical specifications. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the because the failure of a CRDM housing is a Primary System LOCA initiator contributor. The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors

answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a large component rupture. The inspectors determined the finding was indicative of current performance based on the violation documented in section 4OA2 (b.2) of this report. Rather than be duplicative and apply two cross-cutting aspects for the one incident, a cross cutting aspect will not be applied to this issue. Rather it will be applied to the violation documented in 4OA2 (b.2) of this report. (Section 4OA2.b(1))

- Green. The inspectors identified a Green Non-Cited Violation (NCV) of 10 CFR Part 50 Appendix B Criterion V, for failure to follow the root cause procedure. Specifically, the licensee failed to adequately evaluate the generic implications of the cause of the cracking identified in CRDM No. 24.

  The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. The inspectors also answered "yes" to the More-than-Minor screening questions "if left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, absent NRC identification, the licensee would not have completed further evaluations or inspections of CRDM housing welds which could have resulted in additional CRDM housing failure and leakage by TGSCC. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the because the failure of a CRDM housing is a Primary System LOCA initiator contributor. The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a large component rupture. The inspectors determined that the primary cause of the failure to adequately consider welds 3 and 4 on the generic implications section of the root cause report related to the cross-cutting component of Human Performance, Decision Making, because licensee staff did not use conservative assumptions in decision making. Specifically, the licensee did not use conservative assumptions when excluding welds 3 and 4 as being susceptible to TGSCC and therefore include them in the generic implications section of the root cause report. (Item H.1(b)). (Section 4OA2.b(2))

B. Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAILS

4. REACTOR SAFETY

4OA2 Identification and Resolution of Problems (71152)

.3 Selected Issue Follow-up Inspection: Through Wall Leakage of Control Rod Drive Mechanism (CRDM) Housing #24

a. Inspection Scope

On August 12, 2012 the licensee shutdown to investigate an increase in unidentified leakage. The source of the leakage was determined to be a crack in control rod drive mechanism housing (CRDM) No. 24. Shortly after the discovery of the leak in CRDM housing No. 24, the NRC dispatched a special inspection team (SIT) to review the CRDM No. 24 leakage event. The SIT identified an unresolved item (URI) related to the potential failure to prevent recurrence of a significant condition adverse to quality (SCAQ) which was considered an unresolved item, because the licensee's root cause investigation was ongoing at that time. The licensee subsequently removed the failed housing from service for further testing and completed an evaluation to determine the cause of the cracking (CR-PLP-2012-05623).

From March 4, 2013 to March 15, 2013, the inspectors completed one inspection sample regarding problem identification and resolution based upon review of the licensee's root cause report contained in corrective action document CR-PLP-2012-05623.

The inspectors reviewed the licensee's actions in accordance with performance attributes identified in IP 71152. Specifically, the inspectors reviewed licensee corrective action records to determine if: (1) the problems were accurately identified; (2) operability and reportability were adequately ascertained; (3) extent of condition and generic implications were appropriately addressed; (4) classification and prioritization of problem was commensurate with safety significance; (5) root and contributing causes were identified; (6) corrective actions were appropriately focused to correct problem; and (7) timely corrective actions were completed or proposed commensurate with the safety significance of the issues.

b. Findings

.1 Failure to Prevent Recurrence of a Significant Issue Adverse to Quality

Introduction: The inspectors identified a Green Non-Cited Violation (NCV) of 10 CFR Part 50 Appendix B Criterion XVI, and Technical Specification (TS) 3.4.14 Primary Coolant System (PCS) Operational Leakage, for failure to prevent recurrence of a significant condition adverse to quality resulting in a non-compliance with the TS. Specifically, the licensee failed to include the internal CRDM weld build-up area within the scope of corrective actions taken for a 2001 CRDM housing leakage event (a significant condition adverse to quality) and consequently leakage recurred at the CRDM housing No. 24 in 2012.

Description: On August 12, 2012 Palisades Nuclear Power Station shutdown to investigate an increase in unidentified leakage. During a walk-down performed post

shutdown the licensee discovered the source of the leakage to be a pressure boundary leak from (CRDM) Housing No. 24. After further testing, the licensee determined the leak occurred because of a through-wall flaw adjacent to a weld build up on the interior of the housing (weld 5). Weld 5 is a non pressure boundary weld overlay applied to the inside diameter of the CRDM housing to provide for alignment of the CRDM.

The licensee formed a root cause team (RCT) staffed with several licensee personnel. Various vendor sites also provided input used in the root cause investigation. The root cause investigation was conducted in accordance with site procedure EN-LI-118 "Root Cause Evaluation Process" and was documented in root cause analysis report CR-PLP-2012-05623. In this report, the licensee's RCT determined that the probable cause of the cracking was:

> "Stresses in the weld build up area due to manufacturing irregularities and misalignments between CRD-24 upper housing, support tube, and the associated reactor head penetration/CRDM nozzle. Based on lack of cracking found in the other 8 upper housings tested, the failed CRD-24 upper housing contains an as-yet unidentified additional stress".

The RCT also identified the following contributing cause:

> "Transgranular Stress Corrosion Cracking (TGSCC) initiating within the internal weld build-up material of CRD-24. The through wall crack initiated in the weld material and then propagated through the base metal until a leak developed in the OD witness band region at the base of the inner diameter (ID) weld build up.

This conclusion was based upon destructive and non destructive examinations (NDE) completed on a section of the failed housing which included the through-wall flaw. The RCT also relied upon vendor technical reports assessing the results of the NDE as well as vendor calculations related to the stresses in the CRDM housings.

To determine the extent of condition, the licensee performed ultrasonic (UT) examinations of the weld build up area on 8 additional CRDM housings. The licensee selected these locations based on being in a similar location on the head as CRDM-24, and previous cracking having been identified in some of these locations prior to the replacement of the CRDM upper housings and seal housings. The inspectors concluded that this was an appropriate extent of condition review based upon the cause of the CRDM No. 24 failure identified by the licensee.

In 2001 the licensee discovered a steam leak in the housing of CRDM-21 caused by a through-wall TGSCC at CRDM housing weld No. 3 which was located just below the weld build-up region. This issue was categorized as a significant issue adverse to quality by the licensee (CPAL0102186) and the licensee's root cause evaluation was documented in RCR/C-PAL-01-02186. The licensee concluded that the cracks in CRDM-21 were caused by TGSCC which occurred in areas of heavy grinding or machining tool marks. Specifically, this leak was the result of an inner diameter initiated, axially oriented, transgranular crack in the austenitic stainless steel housing material. The failure analysis performed in response to this event identified both axial and circumferential cracks associated with weld 3. Extent of condition inspections revealed additional, non-through wall cracks associated with weld 3 in 41 of the 44 remaining housings for a total of 42 of 45 housings containing cracks.

In response to the observed cracking, Palisades replaced all 45 CRDM housings with housings thought to be more resistant to cracking. Principle changes included:

a. Elimination of weld number 2

b. Relocation of weld number 3 to a higher location thereby minimizing the deposition of crud in the gap between the weld and the bottom plate of the rack and pinion assembly

c. Reduction in residual stresses and cold work on welds by requiring better surface finishes

d. Use of heat sink welding to reduce ID residual tensile stresses

As indicated above, corrective actions taken in response to the 2001 event were limited to pressure boundary welds. The need for corrective actions related to weld 5 was not considered. To evaluate the effectiveness of these preventative actions from the 2001 CRDM leakage event, the inspectors reviewed the licensee actions to determine if they had been sufficient to eliminate one of the 3 necessary factors to cause TGSCC on the CRDM housings: (1) a susceptible material, (2) a corrosive environment and (3) tensile stress." The inspectors identified that the licensee had failed to eliminate one or more of the necessary factors at the weld build-up area to preclude TGSCC in the replacement housing. Specifically:

- The licensee's 2001 root cause report documented that the weld build-up region is exposed to essentially the same environment as the weld that experienced the cracking (corrosive environment remained unchanged).

- No analysis was completed on the stress conditions for the weld build-up region prior to approving the modified replacement housing design (left residual tensile weld stresses on ID of CRDM surface).

- Fabrication restrictions to prohibit grinding were not applied to the weld build-up region (grinding promotes residual tensile stress state on ID of CRDM surface)

- Machining was performed on the weld build-up areas during the fabrication process in order to achieve the dimensions and geometry specified in the design. This process induced cold work stresses in the weld.

- Material was changed from type 347 to type 316 stainless steel (both materials are essentially equally susceptible to TGSCC).

Based upon the recurrence of through-wall leakage in the CRDM housings that occurred at the weld buildup region of the CRDM housings by TGSCC, the inspectors concluded that the licensee actions were not adequate because the appropriate actions to preclude recurrence were within the licensee's ability to foresee and implement. In 1991, the Fort Calhoun plant had experienced through-wall leakage due to TGSCC at the weld build-up region of their CRDM housings (same housing design) and this operational experience had been reviewed by the licensee and dismissed. In their root cause evaluation, the licensee documented their review of the weld build-up region failure by TGSCC at Fort Calhoun in the spare housing and concluded it would not occur at Palisades. This conclusion was based on the assumption that a higher oxygen environment (more aggressive environment) would exist in the spare Fort Calhoun housings than in the inservice Palisades housings. However the licensee did not have a sufficient basis to confirm this assumption, nor did the licensee perform additional testing to determine if the environment of their inservice housings was sufficiently benign to prevent TGSCC. The licensee's 2012 RCT also reached a similar conclusion and documented that due to organizational/ programmatic weakness the 1991 Fort Calhoun operating experience was not adequately utilized to include inspection of the housing ID weld build-up regions. The inspectors identified that the licensee had missed a key opportunity to implement effective corrective actions that could have prevented recurrence of the 2001 leakage event and elected not to pursue these actions because of the cost. Specifically, in EA-EAR-2001-0426-01 the licensee considered fabricating the replacement housings with Inconel 600 material because it was much more resistant to TGSCC. However, the licensee elected not to fabricate the replacement housings using this material because of the increased cost.

During the special inspection the inspectors also identified an unresolved item for the Technical Specification pressure boundary leak. The licensee determined the leakage commenced on July 14, 2012. The licensee operated in this condition for greater than 6 hours, which is the required shutdown time when pressure boundary leakage exists in the plant.

Based on the information provided above, unresolved items 05000255/2012012-01 "TS for PCS Pressure Boundary Leakage" and 05000255/2012012-03 "Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality," are being closed to the following finding and associated violation.

Analysis: The inspectors determined that the licensee's failure to prevent recurrence of TGSCC of the CRDM housings (a significance condition adverse to quality) that resulted in a TS non-compliance was a performance deficiency that warranted a significance evaluation. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. Specifically the licensee did not limit the likelihood of events that upset plant stability by not taking adequate corrective actions to prevent recurrence of leakage in CRDM housings which represents a pressure boundary leakage and a condition prohibited by the technical specifications. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a large component rupture.

The inspectors determined the finding is indicative of current performance based on the violation documented in section 4OA2 (b.2) of this report. Rather than be duplicative and apply two cross-cutting aspects for the one incident, a cross cutting aspect will not be applied to this issue. Rather it will be applied to the violation documented in 4)A2 (b.2) of this report.

Enforcement: The inspectors identified a NCV of 10 CFR Appendix B Criterion XVI "Corrective Actions", and Technical Specification 3.4.13 "Primary Coolant System Operational Leakage", having a very low safety significance (Green), for failure to prevent the recurrence of leakage in CRDM housings due to TGSCC resulting in the operation of the reactor with pressure boundary leakage for greater than the TS allowed time.

Title 10 CFR Appendix B Criterion XVI requires, in part, that "In the case of significant conditions adverse to quality, the cause of the condition is determined and corrective action taken to preclude repetition."

Technical Specification 3.4.13 PCS Operation Leakage states, in part, "PCS operational Leakage shall be limited to no pressure boundary leakage." Condition B requires the licensee be in Hot Standby in 6 hours and Cold Shutdown in 36 hours when pressure boundary leakage exists.

Contrary to the above, from June 2001 through October 6, 2003, the licensee failed to take adequate corrective actions to prevent recurrence of pressure boundary leakage in CRDM housings due to TGSCC. The leakage of CRDM housing No. 21 identified in 2001 was categorized as a significant condition adverse to quality in accordance with the licensee's corrective action program. The licensee performed a root cause evaluation that determined the cause to be TGSCC. The corrective actions to prevent recurrence included changing the design to reduce stresses in the failed weld, control the surface finish of the pressure retaining welds to reduce potential crack initiation points and the welding process was also changed to reduce the stresses in the weld. These corrective actions were narrowly focused on the pressure retaining welds of the CRDM housings. As a result of the narrow focus of corrective actions, on August 12, 2012 a leak was identified from weld 5, a non-pressure boundary weld, of CRDM housing #24. The cause of this leak was also determined to be TGSCC. The source of the leakage was specifically the weld build up region, which was inappropriately excluded from the scope of corrective actions taken in 2001 to prevent recurrence. The pressure boundary leakage was identified due to an increase in unidentified leakage noted on July 14, 2012. The plant did not enter Hot Standby until August 12, 2012 indicating the licensee operated with pressure boundary leakage for greater than the TS allowed time of 6 hours. The licensee is evaluating the issue to determine what further action need to be taken to address the concern. Because of the very low safety significance and because the licensee entered this issue into their corrective action program (CR-PLP-20136-01134), it is being treated as a NCV consistent with Section 2.3.2 of the Enforcement Policy (NCV 05000255/2013002-xx).

.2 Failure to Adequately Address the Generic Implications of the Cracking identified in CRDM 24

Introduction: The inspectors identified a Green Non-Cited Violation (NCV) of 10 CFR Part 50 Appendix B Criterion V, for failure to follow the root cause procedure. Specifically, the licensee failed to adequately evaluate the generic implications of the cause of the cracking identified in CRDM No. 24.

Description: While reviewing the 2012 root cause report CR-PLP-2013-05623 related to the cracking identified in CRDM No. 24 the inspectors identified that the licensee had not appropriately considered the generic implications of the cracking in the extent of condition review. The licensee's proposed corrective actions narrowly focused on the weld build up region (weld 5), instead of broader actions to ensure other CRDM housing welds were fit for their intended service life.

On March 13, the inspectors requested that the licensee identify the bases for excluding other CRDM housing welds (weld #3 below the weld build up region and weld #4 above the weld build up region) from the scope of planned corrective actions. On March 29, the licensee provided additional information to justify excluding these welds from the scope of the corrective actions. The licensee stated that these actions would produce compressive stresses on the ID of welds 3 and 4 making them immune from cracking. The licensee credited the corrective actions associated with the modifications to the CRDM housing design completed in 2001 as the basis to exclude housing welds No 3 and 4 from additional actions to identify the extent of TGSCC. The corrective actions taken in 2001 included performing heat sink welding, which is a methodology used to reduce the stresses on the inner diameter (ID) of the weld, they also changed the design to reduce design stresses at weld #3 and they specified a smoother surface finish (RMS 125) to reduce potential crack initiation points. The inspectors acknowledged that these actions would reduce the tensile stress at the ID surface and thus reduce the probability of initiating TGSCC.

However, the information provided did not demonstrate that TGSCC would not occur because it did not demonstrate that tensile stress would be eliminated at the ID surface during operation. In particular, repairs completed at the inner surface of weld No. 4, would result in high residual tensile stress at the inside surface of the weld which would promote the initiation of TGSCC. Repairs were also performed on weld No. 3; from the outer diameter (OD) surface of the weld. The licensee believed that the last pass heat sink welding process would be sufficient to ensure residual compressive stress would remain at the ID surface of Weld No. 3 even with repairs to the OD surface. However, the licensee had not completed detailed residual weld stress testing or modeling to confirm this assumption.

The inspectors identified that the three factors required for TGSCC could still be present at the welds 3 and 4 as follows:

- Corrosive environment – Weld 3 would operate in a similar environment as the weld build up region of the CRDM housing. Weld No. 4 would be exposed to a lower operating temperature then the weld build up region, however TGSCC can still occur at 250 degrees Fahrenheit as evidenced by the Palisades previous operating experience with cracking identified in the seal housings that operate at even lower temperatures.

- Susceptible material – Welds 3, 4 and 5 are composed of the same weld filler and base metal materials as the weld buildup region (e.g. weld filler material consistent with the type 316 stainless housing base metal). This material would be equally susceptible to TGSCC, as the type 347 stainless steel and weld filler materials used in the pre-2001 CRDM housing design that developed a through wall leak caused by TGSCC at weld No.3.

- Tensile stresses - While it is assumed that the corrective actions taken in response to the 2001 leak will reduce the potential for tensile stresses to exist on the inner surface of CRDM housings at welds 3 and 4, especially in light of the repairs made to welds 3 and 4, it has not been conclusively demonstrated that these tensile stresses have been eliminated. As such it is not reasonable to conclude that tensile stresses are not present and, therefore, the potential for transgranular stress corrosion cracking has been eliminated.

Although the root cause report discusses manufacturing irregularities and misalignment between CRDM housing 24 and the support tube, seismic supports and the associated reactor head penetration/CRDM nozzle as potential source of stresses leading to cracking, the root cause report also states that "based on the lack of cracking found in the other 8 upper housings tested, the failed CRD-24 upper housing contains an as-yet unidentified additional stress." Because the cause of the additional stress was not identified, the licensee had not established a sufficient basis in the RCR to exclude welds 3 and 4 from the extent of condition review (e.g. potential generic implications).

The inspectors identified that the licensee had not followed Procedure EN-LI-118 Root Cause evaluation in the root cause review of the CRDM housing No. 24 leak as documented in report CR-PLP-2013-05623. Section 5.5 (12)e of EN-LI-118 required that the licensee "perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affects other SSC's." In this case, the inspectors identified that the licensee had not documented a sufficient basis in RCR CR-PLP-2013-05623 to exclude welds No. 3 and No. 4 from the generic factors discussed above that led to the 2012 leak in the CRDM housing No. 24 (e.g. TGSCC at the weld buildup region). The licensee entered this issue into the corrective action program as CR-PLP-2013-01500. To restore compliance with the procedure, the licensee intended to revise the inspection plan to add additional corrective actions to inspect a sample of welds No. 3 and 4 for TGSCC during the upcoming refueling outage.

Analysis: The inspectors determined that the failure to adequately evaluate the generic implications of the cause of the cracking identified in CRDM #24 in accordance with the root cause procedure EN-LI-118 was a performance deficiency that warranted a significance evaluation. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. The inspectors also answered "yes" to the More-than-Minor screening questions "if left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, absent NRC identification, the licensee would not have completed further evaluations or inspections of CRDM housing welds which could have resulted in additional CRDM housing failure and leakage by TGSCC. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a large component rupture.

The inspectors determined that the primary cause of the failure to adequately consider welds 3 and 4 on the generic implications section of the root cause report related to the cross-cutting component of Human Performance, Decision Making, because licensee staff did not use conservative assumptions in decision making. Specifically, the licensee did not use conservative assumptions when excluding welds 3 and 4 as being susceptible to TGSCC and therefore include them in the generic implications section of the root cause report. (Item H.1(b) of IMC 310).

Enforcement: The inspectors identified a NCV of 10 CFR Appendix B Criterion V "Instructions, Procedures and Drawings", having a very low safety significance (Green), for failure to adequately evaluate the generic implications of the cause of cracking identified in CRDM #24 as it relates to weld 3 and 4 in accordance with the root cause procedure.

Title 10 CFR Appendix B Criterion V "Instruction, Procedures and Drawings requires in part, "Activities affecting quality shall be prescribed by documented procedures and shall be accomplished in accordance with these procedures."

Procedure EN-LI-118 Root Cause evaluation process revision 17 states:

a. 5.5 (12)e: perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affects other SSC's , organizations or work processes. Use the two step process in accordance with attachment 9.7
b. Attachment 9.7 states Determine whether the occurrence/consequence (problem) is isolated, or whether it has broader (generic or common mode) implications. Achieve this by asking the following questions:
    i. Could this happen to equipment that is similar in function, design, or service condition?
    ii. Could this happen to a group of components? (components of the same construction or materials that could be similarly affected by one condition)
c. Attachment 9.7 also states: Document the results of the above considerations. Include the following items in the write up:

   i. Generic Implications (Is this problem/ cause limited to this component/equipment, or does it apply to others as well)
   ii. Existing broader (generic/common mode) considerations

d. 5.5(15)(10)c&f: Document proposed corrective actions and due dates to address valid generic implications. If no corrective action is recommended for a valid generic implication then document the basis for this conclusion and any risk or consequence identified as a result of taking no action.

Contrary to the above, from February 24, 2013 through April 18, 2013, the licensee failed to perform an activity affecting quality in accordance with procedure EN-LI-118. Specifically, the licensee did not identify and document the existing broader (generic/common mode) considerations associated with TGSCC at CRDM housing welds No. 3 and No. 4. Consequently, the licensee failed to propose corrective actions for the generic implications of TGSCC at CRDM housing welds No. 3 and No. 4. The licensee was considering adding welds 3 and 4 into their inspection plan for activities to be performed during the next refueling outage. Because of the very low safety significance and because the licensee entered this issue into their corrective action program (CR-PLP-2013-01500), it is being treated as a NCV consistent with Section 2.3.2 of the Enforcement Policy (NCV 05000255/2013003-xx).

4OA5 Other Activities

.1 (Closed) Unresolved Item 05000255/2012012-02: Potential Inadequate Degradation Evaluation of CRDM Housings

During a Special Inspection performed in August 2012, NRC inspectors identified an issue which could not be resolved without additional information (Unresolved Issue (URI)). This issue was associated with the rate of growth of the crack which created the through wall leak in CRDM housing 24, discovered on August 12, 2012. Identification of this crack growth rate is significant in determining appropriate intervals for future inspections to provide reasonable assurance that CRDM housing leakage will not recur.

Preliminary failure analysis data available at the time of the inspection indicated that the observed cracking was due to transgranular stress corrosion cracking. Cracking of this type is normally due to the presence of oxygen and chlorides at the location of the crack. When examining the fracture surface at the location the through-wall leak occurred, the licensee identified six concentric rings (beach marks) propagating in a radial direction from the inside diameter out towards the outside diameter of the housing. Beach marks are normally associated with fatigue failures and indicate the number of stress cycles from crack initiation to crack failure. In this case there was no evidence that fatigue contributed to the failure. Despite the lack of evidence of fatigue, it was apparent that the crack which resulted in the CRDM housing 24 leak grew in increments. It was not, however, immediately apparent whether the increments were related to oxygen ingress (refueling outages) or temperature/pressure cycles.

At the time of the original inspection, 5 time intervals for through wall crack growth were under consideration. Two were based on literature crack growth data and three were based on interpretations of the beach marks. These time intervals were:

1. Based on literature data, one contractor estimated that a 10% through wall flaw would require 4 years to reach 50% through wall.
2. Based on literature data another contractor estimated the crack growth rate to be 2.1 x 10^{-5} in/hr or 0.18 in/yr. This is approximately three times faster than the crack growth rate proposed in the above mentioned rate.
3. Based on the concept of oxygen ingress at refueling outages 6 cycles of 18 months duration would require 9 years for the crack to grow through wall
4. Based on the concept of temperature/pressure cycles, the plant experienced 6 cold shutdowns in approximately 2 years preceding the crack. This equates to 2 years for the crack to grow through wall.
5. Based on the concept that oxygen is required for crack growth and that oxygen is rapidly purged from the CRDM housings due to leakage past the seals, crack growth occurs only during the first few weeks of operation following a refueling outage, followed by no growth for the remaining period of operation when oxygen concentrations are low. This equates to 6 oxygen ingress events (irrespective of time between events) for the crack to grow through wall.

NRC inspectors including technical experts from NRC Headquarters performed a follow-up inspection to determine if the assumptions made by the licensee were conservative and the planned actions bounded those conservative assumptions. The inspectors reviewed a variety of documents associated with crack growth and inspection intervals. The inspectors noted various the following statements included in the root cause report and vendor documents related to the determination of the appropriate crack growth rate.

1. The laboratory conducting the failure analysis concluded, it could not be conclusively determined if the beach marks corresponded to refueling outages, (i.e., 18 month cycle) or shorter periods as occurred during outages over the past 24 months
2. Palisades CRDM housing 21 leaked at weld 3 in 2001. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 failure. In calculating the crack growth rate of this crack, one contractor utilized an interval between beach marks which is much shorter than refueling outages. The intervals used are consistent with plant thermal cycles in which oxygen may or may not have been admitted into the CRDMs.
3. A spare CRDM housing at Ft Calhoun leaked at weld 5 in 1990. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 failure. In calculating the crack growth rate of this crack, Ft Calhoun stated that the beach marks were related to refueling cycles. Ft Calhoun also performed calculations indicating that the oxygen level at the location of the flaw did not change with time (including in response to refueling outages) because the spare CRDM housing was not vented. Ft Calhoun's evaluation indicated that oxygen levels at the vicinity of the crack would have begun to decline through diffusion and convection had the intervals between outages been much longer than 18 months. This is interpreted to mean that the beach marks at Ft Calhoun are in response to pressure/thermal cycles.
4. In at least one instance Palisades needed to repair the seals on a reactor coolant pump at a time other than an outage. This necessitates draining some of the water from the reactor coolant system and venting (admitting oxygen into) the CRDM

housing. This represents an additional oxygen ingress event not included when determination of time to cracking is based on refueling outages.
5. In its inspection plan Palisades states that it will inspect all CRDM housings over the next 4 refueling outages, i.e., the interval between inspections is 1 refueling outage

Based on the above review, the inspection team notes that there are certain non conservative statements contained in the Root Cause Report and the inspection plan. These include:

1. The crack growth rate based on refueling outages is understated. If oxygen ingress is related to beach marks, given the oxygen ingress event which occurred to repair reactor coolant pump seals, 6 beach marks would occur in a maximum of 5 refueling intervals rather than the 6 refueling intervals that were used to calculate the crack growth rate in the root cause report.
2. The crack growth rate based on heat up and cool down cycles is overstated. The value in the root cause is based on 11 months. While 6 shutdowns did occur at the plant in 11 months several of these events did not result in pressure/temperature changes of the reactor coolant system. The appropriate time frame is 24 months rather than 11.
3. The inspection plan contains a non conservative statement: "However, once the crack has been initiated it propagates over 4 to 5 operating cycles prior to going through wall." While this statement does reflect one of the proposed theories for crack growth, sufficient evidence to demonstrate reasonable assurance that this theory is correct, and thereby overcome the non-conservatism of this statement, does not exist.

Despite the existence of the non conservatisms stated above, the inspectors conclude:

1. Sufficient evidence to conclusively determine the rate of crack growth does, and will not exist.
2. Crack growth based on pressure/temperature cycles is the most conservative of the potential crack growth mechanisms. In the absence of reasonable assurance of the correctness of less conservative mechanisms, through wall crack growth in 2 years must be utilized for regulatory purposes.
3. The licensee has not formally committed to any of the crack growth mechanisms discussed.
4. The licensee's inspection program includes inspections in each of the next 4 outages. This inspection interval, once per outage, bounds all the crack growth mechanisms considered.

The staff finds this approach to inspection to be both acceptable and sufficient justification to close this URI.

4OA6 Management Meetings

.2 Interim Exit Meetings

An interim exit was conducted for:

- The results of the selected issue follow-up inspection, with Mr. X. XXXX, Nuclear Safety Assurance Director on April 18, 2013.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

B. Davis, Engineering Director
O. Gustafson, Licensing Manager
T. Foudy, Engineering Supervisor
B. Williams, Engineer
B. Dotson, Licensing

LIST OF ITEMS OPENED, CLOSED, DISCUSSED

Closed

05000255/2012012-01	URI	TS for PCS Pressure Boundary Leakage
05000255/2012012-02	URI	Potential Inadequate Degradation Evaluation of CRDM Housings
05000255/2012012-03	URI	Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality

Opened and Discussed

None.

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

4OA5 Other Activities

LIST OF ACRONYMS USED

Sanchez Santiago, Elba

From:	Holmberg, Mel
Sent:	Friday, April 19, 2013 10:34 AM
To:	Sanchez Santiago, Elba
Cc:	Alley, David; Hills, David
Subject:	Comments to Palisades Input
Attachments:	Palisades Input to DRP Report 2013 002 msh comments.docx

Elba,

Please take a look at my suggested comments attached. If you agree with them, it may be appropriate to incorporate the ones in your enforcement section of the criterion XVI violation first and bring copies to our 12:30 meeting with Steve Orth.

M

C29

UNITED STATES
NUCLEAR REGULATORY COMMISSION
LISLE, IL 60532-4352

April XX, 2012

MEMORANDUM TO: Thomas Taylor
Senior Resident Inspector
Palisades Nuclear Plant

FROM: David Hills, Chief
Engineering Branch 3
Division of Reactor Safety

SUBJECT: PALISADES NUCLEAR PLANT
DRS INPUT TO INTEGRATED REPORT 05000255/2013002

Enclosed is the report input for the Palisades Nuclear Plant, Inspection Report 05000255/2013002. This report input documents completion of our review of Unresolved Items 05000255/2012012-01, "TS for PCS Pressure Boundary Leakage," 05000255/2012012-02, "Potential Inadequate Degradation Evaluation of CRDM Housings," and 05000255/2012012-03, "Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality." This report also completes the one sample of the Problem Identification and Resolution, Selected Issue Follow-up in accordance with IP 71152. I have reviewed this input to confirm compliance with Inspection Manual Chapter (IMC) 0612 and IMC 305. This input is ready for inclusion into the integrated report and dissemination to the public.

Please input the following post Inspection Data into RPS:

Inspection Procedure	**Procedure Status** – see below: Incomplete, Complete, Complete by reference, Complete-full sample not available, Complete – opportunity to apply procedure not available, Not Applicable.	**Sample Size** – As documented in Scope Section If less than full sample size documented in the report input, the inspector must provide a justification below to enter into RPS and support the procedure status selected
71152	Complete	1

Inspection Report Item and Type (AV, FIN, NCV, URI or VIO)	**Cornerstone** (IE, MS, BI, EP, OR, PR, MISC)	**Cross Cutting Aspect** (H.n(i), P.n(i), S.n(i))	**Responsible Person/Owner**	**Procedure or TI** (71111.07T)	**RPS Branch Code (e.g. closeout responsibility)** EB1 3820 EB2 3870 EB3 3840 PST (RP) 3860 PSB (Safeguards) 3850 OB 3810
NCV-XXX	IE	n/a	E. Sanchez Santiago	71152	3820
NCV-XXX	IE	H.1(b)	E. Sanchez Santiago	71152	3820

Enclosure: Input to Inspection Report 05000255/2013002

cc w/encl: J. Giessner, Chief
C. Hernandez, Site Admin Assistant

CONTACT: E. Sanchez Santiago, DRS
(630) 829-9715

DOCUMENT NAME: G:\DRSIII\DRS\Work in Progress\-Palisades Input to DRP Report 2013 002 URI EMS.docx
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To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	RIII		NRR		RIII		NRR	
NAME	ESanchezSantiago		DAlley		DHills		TLupold	
DATE	4/ /13							

OFFICIAL RECORD COPY

Cover Letter

X Green findings involving a violation were identified. Include the following:

Based on the results of this inspection, two NRC-identified findings of very low safety significance (Green) were identified. These findings were determined to involve a violation of NRC requirements. However, because of the very low safety significance and because the issues were entered into your corrective action program, the NRC is treating the issue as Non-Cited Violation, in accordance with Section 2.3.2 of the NRC Enforcement Policy.

TITLE PAGE

Inspectors: D. Alley, Senior Materials Engineer
E. Sanchez Santiago, Reactor Inspector

SUMMARY OF FINDINGS

A. NRC-Identified and Self-Revealed Findings

Cornerstones: Initiating Events

- Green. The inspectors identified a Green Non-Cited Violation (NCV) of 10 CFR Part 50 Appendix B Criterion XVI, and Technical Specification (TS) 3.4.14 Primary Coolant System (PCS) Operational Leakage, for failure to prevent recurrence of a significant condition adverse to quality resulting in a non-compliance with the TS. Specifically, the licensee failed to include the internal CRDM weld build-up area within the scope of corrective actions taken for a 2001 CRDM housing leakage event (a significant condition adverse to quality) and consequently leakage recurred at the CRDM housing No. 24 in 2012.

  The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. Specifically the licensee did not limit the likelihood of events that upset plant stability by not taking adequate corrective actions to prevent recurrence of leakage in CRDM housings which represents a pressure boundary leakage and a condition prohibited by the technical specifications. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the because the failure of a CRDM housing is a Primary System LOCA initiator contributor. The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors

answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a large component rupture. The inspectors determined the finding was indicative of current performance based on the violation documented in section 4OA2 (b.2) of this report. Rather than be duplicative and apply two cross-cutting aspects for the one incident, a cross cutting aspect will not be applied to this issue. Rather it will be applied to the violation documented in 4OA2 (b.2) of this report. (Section 4OA2.b(1))

- Green. The inspectors identified a Green Non-Cited Violation (NCV) of 10 CFR Part 50 Appendix B Criterion V, for failure to follow the root cause procedure. Specifically, the licensee failed to adequately evaluate the generic implications of the cause of the cracking identified in CRDM No. 24.

  The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. The inspectors also answered "yes" to the More-than-Minor screening questions "if left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, absent NRC identification, the licensee would not have completed further evaluations or inspections of CRDM housing welds which could have resulted in additional CRDM housing failure and leakage by TGSCC. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the because the failure of a CRDM housing is a Primary System LOCA initiator contributor. The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a large component rupture. The inspectors determined that the primary cause of the failure to adequately consider welds 3 and 4 on the generic implications section of the root cause report related to the cross-cutting component of Human Performance, Decision Making, because licensee staff did not use conservative assumptions in decision making. Specifically, the licensee did not use conservative assumptions when excluding welds 3 and 4 as being susceptible to TGSCC and therefore include them in the generic implications section of the root cause report. (Item H.1(b)). (Section 4OA2.b(2))

B. Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAILS

4. REACTOR SAFETY

4OA2 Identification and Resolution of Problems (71152)

.3 Selected Issue Follow-up Inspection: Through Wall Leakage of Control Rod Drive Mechanism (CRDM) Housing #24

a. Inspection Scope

On August 12, 2012 the licensee shutdown to investigate an increase in unidentified leakage. The source of the leakage was determined to be a crack in control rod drive mechanism housing (CRDM) No. 24. Shortly after the discovery of the leak in CRDM housing No. 24, the NRC dispatched a special inspection team (SIT) to review the CRDM No. 24 leakage event. The SIT identified an unresolved item (URI) related to the potential failure to prevent recurrence of a significant condition adverse to quality (SCAQ) which was considered an unresolved item, because the licensee's root cause investigation was ongoing at that time. The licensee subsequently removed the failed housing from service for further testing and completed an evaluation to determine the cause of the cracking (CR-PLP-2012-05623).

From March 4, 2013 to March 15, 2013, the inspectors completed one inspection sample regarding problem identification and resolution based upon review of the licensee's root cause report contained in corrective action document CR-PLP-2012-05623.

The inspectors reviewed the licensee's actions in accordance with performance attributes identified in IP 71152. Specifically, the inspectors reviewed licensee corrective action records to determine if: (1) the problems were accurately identified; (2) operability and reportability were adequately ascertained; (3) extent of condition and generic implications were appropriately addressed; (4) classification and prioritization of problem was commensurate with safety significance; (5) root and contributing causes were identified; (6) corrective actions were appropriately focused to correct problem; and (7) timely corrective actions were completed or proposed commensurate with the safety significance of the issues.

b. Findings

.1 Failure to Prevent Recurrence of a Significant Issue Adverse to Quality

Introduction: The inspectors identified a Green Non-Cited Violation (NCV) of 10 CFR Part 50 Appendix B Criterion XVI, and Technical Specification (TS) 3.4.14 Primary Coolant System (PCS) Operational Leakage, for failure to prevent recurrence of a significant condition adverse to quality resulting in a non-compliance with the TS. Specifically, the licensee failed to include the internal CRDM weld build-up area within the scope of corrective actions taken for a 2001 CRDM housing leakage event (a significant condition adverse to quality) and consequently leakage recurred at the CRDM housing No. 24 in 2012.

Description: On August 12, 2012 Palisades Nuclear Power Station shutdown to investigate an increase in unidentified leakage. During a walk-down performed post

shutdown the licensee discovered the source of the leakage to be a pressure boundary leak from (CRDM) Housing No. 24. After further testing, the licensee determined the leak occurred because of a through-wall flaw adjacent to a weld build up on the interior of the housing (weld 5). Weld 5 ~~is a~~ ~~non pressure boundary~~ consists of a ~~weld overlay~~ weld material deposit applied to the inside diameter of the CRDM housing which ~~to~~ provides for alignment of the CRDM.

The licensee formed a root cause team (RCT) staffed with ~~several~~ licensee personnel and augmented with input from~~. Various~~ vendors ~~sites also provided input used in the root cause investigation~~. The root cause investigation was conducted in accordance with site procedure EN-LI-118 "Root Cause Evaluation Process" and was documented in root cause analysis report CR-PLP-2012-05623. In this report, the licensee's RCT determined that the probable cause of the cracking was:

> "Stresses in the weld build up area due to manufacturing irregularities and misalignments between CRD-24 upper housing, support tube, and the associated reactor head penetration/CRDM nozzle. Based on lack of cracking found in the other 8 upper housings tested, the failed CRD-24 upper housing contains an as-yet unidentified additional stress".

The RCT also identified the following contributing cause:

> "Transgranular Stress Corrosion Cracking (TGSCC) initiating within the internal weld build-up material of CRD-24. The through wall crack initiated in the weld material and then propagated through the base metal until a leak developed in the OD witness band region at the base of the inner diameter (ID) weld build up.

This conclusion was based upon destructive and non destructive examinations (NDE) completed on a section of the failed housing which included the through-wall flaw. The RCT also relied upon vendor technical reports assessing the results of the NDE as well as vendor calculations related to the stresses in the CRDM housings.

To determine the extent of condition, the licensee performed ultrasonic (UT) examinations of the weld build up area on 8 additional CRDM housings. The licensee selected these locations based on being in a similar location on the head as CRDM-24, and previous cracking having been identified in some of these locations prior to the replacement of the CRDM upper housings and seal housings. Additionally, the licensee was planning to conduct examinations of additional housings during the next refueling outage. The inspectors concluded that this was an appropriate initial extent of condition review based upon the cause of the CRDM No. 24 failure identified by the licensee.

In 2001, the licensee discovered a steam leak in the housing of CRDM-21 caused by a through-wall TGSCC at CRDM housing weld No. 3 which was located just below the weld build-up region. This issue was categorized as a significant issue adverse to quality (SCAQ) by the licensee (CPAL0102186) and the licensee's root cause evaluation was documented in RCR/C-PAL-01-02186. The licensee considered this issue a SCAQ because it met their procedure xxx definition which stated yyy. The licensee concluded that the cracks in CRDM-21 were caused by TGSCC which occurred in areas of heavy grinding or machining tool marks. Specifically, this leak was the result of an inner diameter initiated, axially oriented, transgranular crack in the austenitic stainless steel housing material. The failure analysis performed in response to this event identified both axial and circumferential cracks associated with weld 3. Extent of condition inspections revealed additional, non-through wall cracks associated with weld 3 in 41 of the 44 remaining housings for a total of 42 of 45 housings containing cracks.

In response to the 2001~~observed~~ cracking, Palisades replaced all 45 CRDM housings with housings thought to be more resistant to cracking. Principle changes included:

a. Elimination of weld number 2,

b. Relocation of weld number 3 to a higher location thereby minimizing the deposition of crud in the gap between the weld and the bottom plate of the rack and pinion assembly,

c. Reduction in residual stresses and cold work on welds by requiring better surface finishes, and

d. Use of heat sink welding to reduce ID residual tensile stresses.

~~As indicated above,~~ Licensee corrective actions taken in response to the 2001 event were limited to pressure boundary welds and did not include~~. The need for corrective actions related to~~ weld ~~5 was not considered~~. ~~To evaluate the effectiveness of these preventative actions from the 2001 CRDM leakage event,~~ the inspectors reviewed the licensee actions to determine if they had been sufficient to eliminate one of the 3 necessary factors to cause TGSCC on the CRDM housings: (1) a susceptible material, (2) a corrosive environment and (3) tensile stress." The inspectors identified that the licensee had failed to eliminate one or more of the necessary factors at the weld build-up area to preclude TGSCC in the replacement housing. Specifically:

- The licensee's 2001 root cause report documented that the weld build-up region is exposed to essentially the same environment as the weld that experienced the cracking (corrosive environment remained unchanged).

- No analysis was completed on the stress conditions for the weld build-up region prior to approving the modified replacement housing design (left residual tensile weld stresses on ID of CRDM surface).

- Fabrication restrictions to prohibit grinding were not applied to the weld build-up region (grinding promotes residual tensile stress state on ID of CRDM surface)

- Machining was performed on the weld build-up areas during the fabrication process in order to achieve the dimensions and geometry specified in the design. This process induced cold work stresses in the weld.

- Material was changed from type 347 to type 316 stainless steel (both materials are essentially equally susceptible to TGSCC).

Based upon the recurrence of through-wall leakage in the CRDM housings that occurred at the weld buildup region of the CRDM housings by TGSCC, the inspectors concluded that the licensee actions were not adequate because the appropriate actions to preclude recurrence were within the licensee's ability to foresee and implement. In 1991, the Fort Calhoun plant had experienced through-wall leakage due to TGSCC at the weld build-up

region of their CRDM housings (same housing design) and this operational experience had been reviewed by the licensee and dismissed. In the licensee's 2001 root cause evaluation, the licensee ~~documented their~~ reviewed ~~of~~ the weld build-up region failure by TGSCC at Fort Calhoun in the spare housing and concluded it would not occur at Palisades. This conclusion was based on the assumption that a higher oxygen environment (more aggressive environment) would exist in the spare Fort Calhoun housings than in the inservice Palisades housings. However the licensee did not ~~have a sufficient basis~~ ~~to~~ confirm this assumption, nor did the licensee perform additional testing to determine if the environment of their inservice housings was sufficiently benign to prevent TGSCC. The licensee's 2012 RCT ~~also~~ reached a similar conclusion and documented that due to organizational/ programmatic weakness at Palisades, the 1991 Fort Calhoun operating experience was not adequately utilized to include inspection of the housing ID weld build-up regions. The inspectors identified that the licensee had missed a key opportunity to implement effective corrective actions that could have prevented recurrence of the 2001 leakage event and elected not to pursue these actions because of the cost. Specifically, in EA-EAR-2001-0426-01 the licensee considered fabricating the replacement housings with Inconel 600 material because it was much more resistant to TGSCC. However, the licensee elected not to fabricate the replacement housings using this material because of the increased cost.

In January of 2002, an NRC special inspection team (SIT) (reference IR 50-2555/01-15) reviewed the licensee corrective actions associated with the through-wall leakage of the CRDM-21 housing caused by TGSCC. The NRC concluded that the licensee design changes in the replacement housings related to CRDM weld number 3 should prevent recurrence of leakage. The NRC conclusion was based the assumption that the licensee would effectively implement the proposed corrective actions, and this did not occur. For example, the NRC had agreed with licensee design changes to relocate weld No. 3 and to use last past heat sink welding to ensure that tensile stress was eliminated at the ID surface to preclude recurrence of TGSCC. However, the licensee allowed weld repairs to OD and ID surfaces of welds No. 3 and 4, which can create tensile stress at the ID surface that promote TGSCC. The NRC SIT also reviewed the licensee's evaluation of the Fort Calhoun leakage caused by through-wall TGSCC the spare CRDM housings. At the time of review, the NRC SIT agreed that the licensee had insufficient information, based upon this event to have reasonably prevented the through-wall leakage at CRDM-21. However, given the Palisades site-specific history of repetitive failures (leakage) of seal housings and CRDM-21 leak caused by TGSCC, the licensee should have (and did not) implemented corrective actions to confirm assumptions related to the operating environment of their housings. At the time of the NRC SIT review, the licensee had assumed that the lower oxygen environment of their CRDM housings would make them less susceptible to TGSCC, but did not followup with measurement of the operating environment (e.g. measure oxygen) to confirm this conclusion, nor did the licensee elect to implement NDE on weld 5 to detect and prevent cracking prior the through-wall leakage in CRDM 24. Therefore, the inspectors concluded that the licensee did not effectively implement corrective actions for the 2001 CRDM housing leak resulting in the 2011 CRDM-24 housing leak.

During the 2012 NRC special inspection the ~~inspectors also~~ NRC identified an unresolved item for the Technical Specification pressure boundary leak. The licensee determined the CRDM-24 leakage commenced on July 14, 2012~~. The licensee~~ and that the plant continued to operate~~d~~ in this condition for greater than 6 hours, which ~~is~~ was greater than the required shutdown time with ~~when~~ pressure boundary leakage per TS

LCO xx. ~~exists in the plant.~~ Based on the ~~information provided~~ review discussed above, unresolved items 05000255/2012012-01 "TS for PCS Pressure Boundary Leakage" and 05000255/2012012-03 "Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality," are closed. ~~being closed to the following finding and associated violation~~.

Analysis: The inspectors determined that the licensee's failure to prevent recurrence of TGSCC of the CRDM housings (a significance condition adverse to quality) that resulted in a TS non-compliance was a performance deficiency that warranted a significance evaluation. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. Specifically the licensee did not limit the likelihood of events that upset plant stability by not taking adequate corrective actions to prevent recurrence of leakage in CRDM housings which represents a pressure boundary leakage and a condition prohibited by the Technical Specifications. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a ~~large~~ component rupture.

The inspectors determined this finding was caused by the same errors that led to the violation discussed in section 4OA2 (b.2) of this report and is indicative of current performance. ~~based on the violation documented in section 4OA2 (b.2) of this report~~. Because the very similar cause for this performance deficiency and the one discussed in Section 4OA2 (b.2) of this report, no separate cross-cutting aspect is assigned to this finding. ~~is the same as the cause of the perRather than be duplicative and apply two cross-cutting aspects for the one incident, a cross cutting aspect will not be applied to this issue. Rather it will be applied to the violation documented in 4)A2 (b.2) of this report.~~

Enforcement: The inspectors identified a NCV of 10 CFR Appendix B Criterion XVI "Corrective Actions", and Technical Specification 3.4.13 "Primary Coolant System Operational Leakage", having a very low safety significance (Green), for failure to prevent the recurrence of leakage in CRDM housings due to TGSCC resulting in the operation of the reactor with pressure boundary leakage for greater than the TS allowed time.

Title 10 CFR Appendix B Criterion XVI requires, in part, that "In the case of significant conditions adverse to quality, the cause of the condition is determined and corrective action taken to preclude repetition."

Technical Specification 3.4.13 PCS Operation Leakage states, in part, "PCS operational Leakage shall be limited to no pressure boundary leakage." Condition B requires the licensee be in Hot Standby in 6 hours and Cold Shutdown in 36 hours when pressure boundary leakage exists.

Contrary to the above, from June 2001 through October 6, 2003, the licensee failed to take ~~adequate~~ corrective actions to prevent recurrence of pressure boundary leakage in CRDM housings due to TGSCC. Specifically, the licensee failed to implement corrective actions to preclude the use of materials in an environment known to promote TGSCC or to implement an inspection program that would detect TGSCC in the CRDM housings prior to pressure boundary leakage~~. that would would or implement The leakage of CRDM housing No. 21 identified in 2001 was categorized as a significant condition adverse to quality in accordance with the licensee's corrective action program. The licensee performed a root cause evaluation that determined the cause to be TGSCC. The corrective actions to prevent recurrence included changing the design to reduce stresses in the failed weld, control the surface finish of the pressure retaining welds to reduce potential crack initiation points and the welding process was also changed to reduce the stresses in the weld.~~ Because, licensee ~~These~~ corrective actions were too narrow ~~ly focused on the pressure retaining welds of the CRDM housings. As a result of the narrow focus of corrective actions, on~~ on August 12, 2012 a leak ~~was identified~~ developed ~~from~~ at weld 5, ~~a non-pressure boundary weld, of~~ on CRDM housing #24~~.~~ ~~The~~ caused by ~~of this leak was also determined to be~~ TGSCC. ~~The source of the leakage was specifically the weld build up region, which was inappropriately excluded from the scope of corrective actions taken in 2001 to prevent recurrence.~~ The pressure boundary leakage at CRDM 24 ~~was identified due to an increase in unidentified leakage noted on~~ began on July 14, 2012 and the plant continued to operate until~~. The plant did not enter Hot Standby until~~ August 12, 2012 ~~indicating the licensee operated with~~ which exceeded the 6 hours ~~pressure boundary leakage~~ allowed by TS 3.4.13. ~~for greater than the TS allowed time of 6 hours.~~ At the conclusion of the inspection, the licensee was ~~is evaluating the issue~~ considering an augmented inspection program to detect TGSCC of the CRDM housings that included welds 3,4 and 5~~. to determine what further action need to be taken to address the concern.~~ Because of the very low safety significance and because the licensee entered this issue into their corrective action program (CR-PLP-20136-01134), it is being treated as a NCV consistent with Section 2.3.2 of the Enforcement Policy (NCV 05000255/2013002-xx).

.2 <u>Failure to Adequately Address the Generic Implications of the Cracking identified in CRDM 24</u>

<u>Introduction:</u> The inspectors identified a Green Non-Cited Violation (NCV) of 10 CFR Part 50 Appendix B Criterion V, for failure to follow the root cause procedure. Specifically, the licensee failed to adequately evaluate the generic implications of the cause of the cracking identified in CRDM No. 24.

Description: While reviewing the 2012 root cause report CR-PLP-2013-05623 related to the cracking identified in CRDM No. 24 the inspectors identified that the licensee had not appropriately considered the generic implications of the cracking in the extent of condition review. The licensee's proposed corrective actions narrowly focused on the weld build up region (weld 5), instead of broader actions to ensure other CRDM housing welds were fit for their intended service life.

On March 13, the inspectors requested that the licensee identify the bases for excluding other CRDM housing welds (weld #3 below the weld build up region and weld #4 above the weld build up region) from the scope of planned corrective actions. On March 29, the licensee provided additional information to justify excluding these welds from the scope of the corrective actions. The licensee stated that these actions would produce compressive stresses on the ID of welds 3 and 4 making them immune from cracking. The licensee credited the corrective actions associated with the modifications to the CRDM housing design completed in 2001 as the basis to exclude housing welds No 3 and 4 from additional actions to identify the extent of TGSCC. The corrective actions taken in 2001 included performing heat sink welding, which is a methodology used to reduce the stresses on the inner diameter (ID) of the weld, they also changed the design to reduce design stresses at weld #3 and they specified a smoother surface finish (RMS 125) to reduce potential crack initiation points. The inspectors acknowledged that these actions would reduce the tensile stress at the ID surface and thus reduce the probability of initiating TGSCC.

However, the information provided did not demonstrate that TGSCC would not occur because it did not demonstrate that tensile stress would be eliminated at the ID surface during operation. In particular, repairs completed at the inner surface of weld No. 4, would result in high residual tensile stress at the inside surface of the weld which would promote the initiation of TGSCC. Repairs were also performed on weld No. 3; from the outer diameter (OD) surface of the weld. The licensee believed that the last pass heat sink welding process would be sufficient to ensure residual compressive stress would remain at the ID surface of Weld No. 3 even with repairs to the OD surface. However, the licensee had not completed detailed residual weld stress testing or modeling to confirm this assumption.

The inspectors identified that the three factors required for TGSCC could still be present at the welds 3 and 4 as follows:

- Corrosive environment – Weld 3 would operate in a similar environment as the weld build up region of the CRDM housing. Weld No. 4 would be exposed to a lower operating temperature then the weld build up region, however TGSCC can still occur at 250 degrees Fahrenheit as evidenced by the Palisades previous operating experience with cracking identified in the seal housings that operate at even lower temperatures.

- Susceptible material – Welds 3, 4 and 5 are composed of the same weld filler and base metal materials as the weld buildup region (e.g. weld filler material consistent with the type 316 stainless housing base metal). This material would be equally susceptible to TGSCC, as the type 347 stainless steel and weld filler materials used in the pre-2001 CRDM housing design that developed a through wall leak caused by TGSCC at weld No.3.

- Tensile stresses - While it is assumed that the corrective actions taken in response to the 2001 leak will reduce the potential for tensile stresses to exist on the inner surface of CRDM housings at welds 3 and 4, especially in light of the repairs made to welds 3 and 4, it has not been conclusively demonstrated that these tensile stresses have been eliminated. As such it is not reasonable to conclude that tensile stresses are not present and, therefore, the potential for transgranular stress corrosion cracking has been eliminated.

Although the root cause report discusses manufacturing irregularities and misalignment between CRDM housing 24 and the support tube, seismic supports and the associated reactor head penetration/CRDM nozzle as potential source of stresses leading to cracking, the root cause report also states that "based on the lack of cracking found in the other 8 upper housings tested, the failed CRD-24 upper housing contains an as-yet unidentified additional stress." Because the cause of the additional stress was not identified, the licensee had not established a sufficient basis in the RCR to exclude welds 3 and 4 from the extent of condition review (e.g. potential generic implications).

The inspectors identified that the licensee had not followed Procedure EN-LI-118 Root Cause evaluation in the root cause review of the CRDM housing No. 24 leak as documented in report CR-PLP-2013-05623. Section 5.5 (12)e of EN-LI-118 required that the licensee "perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affects other SSC's." In this case, the inspectors identified that the licensee had not documented a sufficient basis in RCR CR-PLP-2013-05623 to exclude welds No. 3 and No. 4 from the generic factors discussed above that led to the 2012 leak in the CRDM housing No. 24 (e.g. TGSCC at the weld buildup region). The licensee entered this issue into the corrective action program as CR-PLP-2013-01500. To restore compliance with the procedure, the licensee intended to revise the inspection plan to add additional corrective actions to inspect a sample of welds No. 3 and 4 for TGSCC during the upcoming refueling outage.

<u>Analysis:</u> The inspectors determined that the failure to adequately evaluate the generic implications of the cause of the cracking identified in CRDM #24 in accordance with the root cause procedure EN-LI-118 was a performance deficiency that warranted a significance evaluation. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. The inspectors also answered "yes" to the More-than-Minor screening questions "if left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, absent NRC identification, the licensee would not have completed further evaluations or inspections of CRDM housing welds which could have resulted in additional CRDM housing failure and leakage by TGSCC. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a large component rupture.

The inspectors determined that the primary cause of the failure to adequately consider welds 3 and 4 on the generic implications section of the root cause report related to the cross-cutting component of Human Performance, Decision Making, because licensee staff did not use conservative assumptions in decision making. Specifically, the licensee did not use conservative assumptions when excluding welds 3 and 4 as being susceptible to TGSCC and therefore include them in the generic implications section of the root cause report. (Item H.1(b) of IMC 310).

Enforcement: The inspectors identified a NCV of 10 CFR Appendix B Criterion V "Instructions, Procedures and Drawings", having a very low safety significance (Green), for failure to adequately evaluate the generic implications of the cause of cracking identified in CRDM #24 as it relates to weld 3 and 4 in accordance with the root cause procedure.

Title 10 CFR Appendix B Criterion V "Instruction, Procedures and Drawings requires in part, "Activities affecting quality shall be prescribed by documented procedures and shall be accomplished in accordance with these procedures."

Procedure EN-LI-118 Root Cause evaluation process revision 17 states:

a. 5.5 (12)e: perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affects other SSC's , organizations or work processes. Use the two step process in accordance with attachment 9.7
b. Attachment 9.7 states Determine whether the occurrence/consequence (problem) is isolated, or whether it has broader (generic or common mode) implications. Achieve this by asking the following questions:
    i. Could this happen to equipment that is similar in function, design, or service condition?
    ii. Could this happen to a group of components? (components of the same construction or materials that could be similarly affected by one condition)
c. Attachment 9.7 also states: Document the results of the above considerations. Include the following items in the write up:
    i. Generic Implications (Is this problem/ cause limited to this component/equipment, or does it apply to others as well)
    ii. Existing broader (generic/common mode) considerations
d. 5.5(15)(10)c&f: Document proposed corrective actions and due dates to address valid generic implications. If no corrective action is recommended for a valid generic implication then document the basis for this conclusion and any risk or consequence identified as a result of taking no action.

Contrary to the above, from February 24, 2013 through April 18, 2013, the licensee failed to perform an activity affecting quality in accordance with procedure EN-LI-118. Specifically, the licensee did not identify and document the existing broader (generic/common mode) considerations associated with TGSCC at CRDM housing welds No. 3 and No. 4. Consequently, the licensee failed to propose corrective actions for the generic implications of TGSCC at CRDM housing welds No. 3 and No. 4. The licensee was considering adding welds 3 and 4 into their inspection plan for activities to be performed during the next refueling outage. Because of the very low safety significance and because the licensee entered this issue into their corrective action program (CR-PLP-2013-01500), it is being treated as a NCV consistent with Section 2.3.2 of the Enforcement Policy (NCV 05000255/2013003-xx).

4OA5 Other Activities

.1 (Closed) Unresolved Item 05000255/2012012-02: Potential Inadequate Degradation Evaluation of CRDM Housings

During a Special Inspection performed in August 2012, NRC inspectors identified an issue which could not be resolved without additional information (Unresolved Issue (URI)). This issue was associated with the rate of growth of the crack which created the through wall leak in CRDM housing 24, discovered on August 12, 2012. Identification of this crack growth rate is significant in determining appropriate intervals for future inspections to provide reasonable assurance that CRDM housing leakage will not recur.

Preliminary failure analysis data available at the time of the inspection indicated that the observed cracking was due to transgranular stress corrosion cracking. Cracking of this type is normally due to the presence of oxygen and chlorides at the location of the crack. When examining the fracture surface at the location the through-wall leak occurred, the licensee identified six concentric rings (beach marks) propagating in a radial direction from the inside diameter out towards the outside diameter of the housing. Beach marks are normally associated with fatigue failures and indicate the number of stress cycles from crack initiation to crack failure. In this case there was no evidence that fatigue contributed to the failure. Despite the lack of evidence of fatigue, it was apparent that the crack which resulted in the CRDM housing 24 leak grew in increments. It was not, however, immediately apparent whether the increments were related to oxygen ingress (refueling outages) or temperature/pressure cycles.

At the time of the original inspection, 5 time intervals for through wall crack growth were under consideration. Two were based on literature crack growth data and three were based on interpretations of the beach marks. These time intervals were:

1. Based on literature data, one contractor estimated that a 10% through wall flaw would require 4 years to reach 50% through wall.
2. Based on literature data another contractor estimated the crack growth rate to be 2.1 x 10^{-5} in/hr or 0.18 in/yr. This is approximately three times faster than the crack growth rate proposed in the above mentioned rate.

3. Based on the concept of oxygen ingress at refueling outages 6 cycles of 18 months duration would require 9 years for the crack to grow through wall
4. Based on the concept of temperature/pressure cycles, the plant experienced 6 cold shutdowns in approximately 2 years preceding the crack. This equates to 2 years for the crack to grow through wall.
5. Based on the concept that oxygen is required for crack growth and that oxygen is rapidly purged from the CRDM housings due to leakage past the seals, crack growth occurs only during the first few weeks of operation following a refueling outage, followed by no growth for the remaining period of operation when oxygen concentrations are low. This equates to 6 oxygen ingress events (irrespective of time between events) for the crack to grow through wall.

NRC inspectors including technical experts from NRC Headquarters performed a follow-up inspection to determine if the assumptions made by the licensee were conservative and the planned actions bounded those conservative assumptions. The inspectors reviewed a variety of documents associated with crack growth and inspection intervals. The inspectors noted various the following statements included in the root cause report and vendor documents related to the determination of the appropriate crack growth rate.

1. The laboratory conducting the failure analysis concluded, it could not be conclusively determined if the beach marks corresponded to refueling outages, (i.e., 18 month cycle) or shorter periods as occurred during outages over the past 24 months
2. Palisades CRDM housing 21 leaked at weld 3 in 2001. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 failure. In calculating the crack growth rate of this crack, one contractor utilized an interval between beach marks which is much shorter than refueling outages. The intervals used are consistent with plant thermal cycles in which oxygen may or may not have been admitted into the CRDMs.
3. A spare CRDM housing at Ft Calhoun leaked at weld 5 in 1990. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 failure. In calculating the crack growth rate of this crack, Ft Calhoun stated that the beach marks were related to refueling cycles. Ft Calhoun also performed calculations indicating that the oxygen level at the location of the flaw did not change with time (including in response to refueling outages) because the spare CRDM housing was not vented. Ft Calhoun's evaluation indicated that oxygen levels at the vicinity of the crack would have begun to decline through diffusion and convection had the intervals between outages been much longer than 18 months. This is interpreted to mean that the beach marks at Ft Calhoun are in response to pressure/thermal cycles.
4. In at least one instance Palisades needed to repair the seals on a reactor coolant pump at a time other than an outage. This necessitates draining some of the water from the reactor coolant system and venting (admitting oxygen into) the CRDM housing. This represents an additional oxygen ingress event not included when determination of time to cracking is based on refueling outages.
5. In its inspection plan Palisades states that it will inspect all CRDM housings over the next 4 refueling outages, i.e., the interval between inspections is 1 refueling outage

Based on the above review, the inspection team notes that there are certain non conservative statements contained in the Root Cause Report and the inspection plan. These include:

1. The crack growth rate based on refueling outages is understated. If oxygen ingress is related to beach marks, given the oxygen ingress event which occurred to repair reactor coolant pump seals, 6 beach marks would occur in a maximum of 5 refueling intervals rather than the 6 refueling intervals that were used to calculate the crack growth rate in the root cause report.
2. The crack growth rate based on heat up and cool down cycles is overstated. The value in the root cause is based on 11 months. While 6 shutdowns did occur at the plant in 11 months several of these events did not result in pressure/temperature changes of the reactor coolant system. The appropriate time frame is 24 months rather than 11.
3. The inspection plan contains a non conservative statement: "However, once the crack has been initiated it propagates over 4 to 5 operating cycles prior to going through wall." While this statement does reflect one of the proposed theories for crack growth, sufficient evidence to demonstrate reasonable assurance that this theory is correct, and thereby overcome the non-conservatism of this statement, does not exist.

Despite the existence of the non conservatisms stated above, the inspectors conclude:

1. Sufficient evidence to conclusively determine the rate of crack growth does, and will not exist.
2. Crack growth based on pressure/temperature cycles is the most conservative of the potential crack growth mechanisms. In the absence of reasonable assurance of the correctness of less conservative mechanisms, through wall crack growth in 2 years must be utilized for regulatory purposes.
3. The licensee has not formally committed to any of the crack growth mechanisms discussed.
4. The licensee's inspection program includes inspections in each of the next 4 outages. This inspection interval, once per outage, bounds all the crack growth mechanisms considered.

The staff finds this approach to inspection to be both acceptable and sufficient justification to close this URI.

4OA6 Management Meetings

.2 Interim Exit Meetings

An interim exit was conducted for:

- The results of the selected issue follow-up inspection, with Mr. X. XXXX, Nuclear Safety Assurance Director on April 18, 2013.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

B. Davis, Engineering Director
O. Gustafson, Licensing Manager
T. Foudy, Engineering Supervisor
B. Williams, Engineer
B. Dotson, Licensing

LIST OF ITEMS OPENED, CLOSED, DISCUSSED

Closed

05000255/2012012-01	URI	TS for PCS Pressure Boundary Leakage
05000255/2012012-02	URI	Potential Inadequate Degradation Evaluation of CRDM Housings
05000255/2012012-03	URI	Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality

Opened and Discussed

None.

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

4OA5 Other Activities

LIST OF ACRONYMS USED

Release entirety

Giessner, John

From: Sanchez Santiago, Elba
Sent: Monday, April 22, 2013 5:04 PM
To: Giessner, John
Subject: FW: Palisades Report
Attachments: Palisades Input to DRP Report 2013 002 URI EMS.docx

Follow Up Flag: Follow up
Flag Status: Completed

Jack,

Attached is the input to the Palisades report for your review. Let me know if you have any comments or questions.

Thanks,
Elba

From: Sanchez Santiago, Elba
Sent: Monday, April 22, 2013 4:24 PM
To: Hills, David
Cc: Holmberg, Mel
Subject: Palisades Report

Dave,

Attached is the palisades inspection report input including the changes proposed on Friday. Let me know if you have any comments or questions.

Thanks,
Elba

C30

UNITED STATES
NUCLEAR REGULATORY COMMISSION
LISLE, IL 60532-4352

April XX, 2012

MEMORANDUM TO: Thomas Taylor
Senior Resident Inspector
Palisades Nuclear Plant

FROM: David Hills, Chief
Engineering Branch 3
Division of Reactor Safety

SUBJECT: PALISADES NUCLEAR PLANT
DRS INPUT TO INTEGRATED REPORT 05000255/2013002

Enclosed is the report input for the Palisades Nuclear Plant, Inspection Report 05000255/2013002. This report input documents completion of our review of Unresolved Items 05000255/2012012-01, "TS for PCS Pressure Boundary Leakage," 05000255/2012012-02, "Potential Inadequate Degradation Evaluation of CRDM Housings," and 05000255/2012012-03, "Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality." This report also completes the one sample of the Problem Identification and Resolution, Selected Issue Follow-up in accordance with IP 71152. I have reviewed this input to confirm compliance with Inspection Manual Chapter (IMC) 0612 and IMC 305. This input is ready for inclusion into the integrated report and dissemination to the public.

Please input the following post Inspection Data into RPS:

Inspection Procedure	**Procedure Status** – see below: Incomplete, Complete, Complete by reference, Complete-full sample not available, Complete – opportunity to apply procedure not available, Not Applicable.	**Sample Size** – As documented in Scope Section If less than full sample size documented in the report input, the inspector must provide a justification below to enter into RPS and support the procedure status selected
71152	Complete	1

Inspection Report Item and Type (AV, FIN, NCV, URI or VIO)	**Cornerstone** (IE, MS, BI, EP, OR, PR, MISC)	**Cross Cutting Aspect** (H.n(i), P.n(i), S.n(i))	**Responsible Person/Owner**	**Procedure or TI** (71111.07T)	**RPS Branch Code (e.g. closeout responsibility)** EB1 3820, EB2 3870, EB3 3840, PST (RP) 3860, PSB (Safeguards) 3850, OB 3810
NCV-XXX	IE	n/a	E. Sanchez Santiago	71152	3820
NCV-XXX	IE	H.1(b)	E. Sanchez Santiago	71152	3820

Enclosure: Input to Inspection Report 05000255/2013002

cc w/encl: J. Giessner, Chief
C. Hernandez, Site Admin Assistant

CONTACT: E. Sanchez Santiago, DRS
(630) 829-9715

DOCUMENT NAME: G:\DRSIII\DRS\Work in Progress\-Palisades Input to DRP Report 2013 002 URI EMS.docx
9 Publicly Available 9 Non-Publicly Available 9 Sensitive 9 Non-Sensitive
To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	RIII		NRR		RIII		NRR	
NAME	ESanchezSantiago		DAlley		DHills		TLupold	
DATE	4/ /13							

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Cover Letter

X Green findings involving a violation were identified. Include the following:

Based on the results of this inspection, two NRC-identified findings of very low safety significance (Green) were identified. These findings were determined to involve a violation of NRC requirements. However, because of the very low safety significance and because the issues were entered into your corrective action program, the NRC is treating the issue as Non-Cited Violation, in accordance with Section 2.3.2 of the NRC Enforcement Policy.

TITLE PAGE

Inspectors: D. Alley, Senior Materials Engineer
E. Sanchez Santiago, Reactor Inspector

SUMMARY OF FINDINGS

A. NRC-Identified and Self-Revealed Findings

Cornerstones: Initiating Events

- Green. The inspectors identified a Green Non-Cited Violation (NCV) of 10 CFR Part 50 Appendix B Criterion XVI, and Technical Specification (TS) 3.4.14 Primary Coolant System (PCS) Operational Leakage, for failure to prevent recurrence of a significant condition adverse to quality resulting in a non-compliance with the TS. Specifically, the licensee failed to include the internal CRDM weld build-up area within the scope of corrective actions taken for a 2001 CRDM housing leakage event (a significant condition adverse to quality) and consequently leakage recurred at the CRDM housing No. 24 in 2012.

  The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. Specifically the licensee did not limit the likelihood of events that upset plant stability by not taking adequate corrective actions to prevent recurrence of leakage in CRDM housings which represents a pressure boundary leakage and a condition prohibited by the technical specifications. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the because the failure of a CRDM housing is a Primary System LOCA initiator contributor. The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors

answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a large component rupture. The inspectors determined the finding was indicative of current performance based on the violation documented in section 4OA2 (b.2) of this report. Rather than be duplicative and apply two cross-cutting aspects for the one incident, a cross cutting aspect will not be applied to this issue. Rather it will be applied to the violation documented in 4OA2 (b.2) of this report. (Section 4OA2.b(1))

- Green. The inspectors identified a Green Non-Cited Violation (NCV) of 10 CFR Part 50 Appendix B Criterion V, for failure to follow the root cause procedure. Specifically, the licensee failed to adequately evaluate the generic implications of the cause of the cracking identified in CRDM No. 24.

  The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. The inspectors also answered "yes" to the More-than-Minor screening questions "if left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, absent NRC identification, the licensee would not have completed further evaluations or inspections of CRDM housing welds which could have resulted in additional CRDM housing failure and leakage by TGSCC. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the because the failure of a CRDM housing is a Primary System LOCA initiator contributor. The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a large component rupture. The inspectors determined that the primary cause of the failure to adequately consider welds 3 and 4 on the generic implications section of the root cause report related to the cross-cutting component of Human Performance, Decision Making, because licensee staff did not use conservative assumptions in decision making. Specifically, the licensee did not use conservative assumptions when excluding welds 3 and 4 as being susceptible to TGSCC and therefore include them in the generic implications section of the root cause report. (Item H.1(b)). (Section 4OA2.b(2))

B. Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAILS

4. REACTOR SAFETY

4OA2 Identification and Resolution of Problems (71152)

.3 Selected Issue Follow-up Inspection: Through Wall Leakage of Control Rod Drive Mechanism (CRDM) Housing #24

a. Inspection Scope

On August 12, 2012 the licensee shutdown to investigate an increase in unidentified leakage. The source of the leakage was determined to be a crack in control rod drive mechanism housing (CRDM) No. 24. Shortly after the discovery of the leak in CRDM housing No. 24, the NRC dispatched a special inspection team (SIT) to review the CRDM No. 24 leakage event. The SIT identified an unresolved item (URI) related to the potential failure to prevent recurrence of a significant condition adverse to quality (SCAQ) which was considered an unresolved item, because the licensee's root cause investigation was ongoing at that time. The licensee subsequently removed the failed housing from service for further testing and completed an evaluation to determine the cause of the cracking (CR-PLP-2012-05623).

From March 4, 2013 to March 15, 2013, the inspectors completed one inspection sample regarding problem identification and resolution based upon review of the licensee's root cause report contained in corrective action document CR-PLP-2012-05623.

The inspectors reviewed the licensee's actions in accordance with performance attributes identified in IP 71152. Specifically, the inspectors reviewed licensee corrective action records to determine if: (1) the problems were accurately identified; (2) operability and reportability were adequately ascertained; (3) extent of condition and generic implications were appropriately addressed; (4) classification and prioritization of problem was commensurate with safety significance; (5) root and contributing causes were identified; (6) corrective actions were appropriately focused to correct problem; and (7) timely corrective actions were completed or proposed commensurate with the safety significance of the issues.

b. Findings

.1 Failure to Prevent Recurrence of a Significant Issue Adverse to Quality

Introduction: The inspectors identified a Green Non-Cited Violation (NCV) of 10 CFR Part 50 Appendix B Criterion XVI, and Technical Specification (TS) 3.4.14 Primary Coolant System (PCS) Operational Leakage, for failure to prevent recurrence of a significant condition adverse to quality resulting in a non-compliance with the TS. Specifically, the licensee failed to include the internal CRDM weld build-up area within the scope of corrective actions taken for a 2001 CRDM housing leakage event (a significant condition adverse to quality) and consequently leakage recurred at the CRDM housing No. 24 in 2012.

Description: On August 12, 2012 Palisades Nuclear Power Station shutdown to investigate an increase in unidentified leakage. During a walk-down performed post

shutdown the licensee discovered the source of the leakage to be a pressure boundary leak from (CRDM) Housing No. 24. After further testing, the licensee determined the leak occurred because of a through-wall flaw adjacent to a weld build up on the interior of the housing (weld 5). Weld 5 consists of a weld material deposit applied to the inside diameter of the CRDM housing which provides for alignment of the CRDM.

The licensee formed a root cause team (RCT) staffed with licensee personnel and augmented with input from vendors. The root cause investigation was conducted in accordance with site procedure EN-LI-118 "Root Cause Evaluation Process" and was documented in root cause analysis report CR-PLP-2012-05623. In this report, the licensee's RCT determined that the probable cause of the cracking was:

> "Stresses in the weld build up area due to manufacturing irregularities and misalignments between CRD-24 upper housing, support tube, and the associated reactor head penetration/CRDM nozzle. Based on lack of cracking found in the other 8 upper housings tested, the failed CRD-24 upper housing contains an as-yet unidentified additional stress".

The RCT also identified the following contributing cause:

> "Transgranular Stress Corrosion Cracking (TGSCC) initiating within the internal weld build-up material of CRD-24. The through wall crack initiated in the weld material and then propagated through the base metal until a leak developed in the OD witness band region at the base of the inner diameter (ID) weld build up.

This conclusion was based upon destructive and non destructive examinations (NDE) completed on a section of the failed housing which included the through-wall flaw. The RCT also relied upon vendor technical reports assessing the results of the NDE as well as vendor calculations related to the stresses in the CRDM housings.

To determine the extent of condition, the licensee performed ultrasonic (UT) examinations of the weld build up area on 8 additional CRDM housings. The licensee selected these locations based on being in a similar location on the head as CRDM-24, and previous cracking having been identified in some of these locations prior to the replacement of the CRDM upper housings and seal housings. Additionally, the licensee was planning to conduct examinations of additional housings during the next refueling outage. The inspectors concluded that this was an appropriate initial extent of condition review based upon the cause of the CRDM No. 24 failure identified by the licensee.

In 2001, the licensee discovered a steam leak in the housing of CRDM-21 caused by a through-wall TGSCC at CRDM housing weld No. 3 which was located just below the weld build-up region. This issue was categorized as a significant issue adverse to quality (SCAQ) by the licensee (CPAL0102186) and the licensee's root cause evaluation was documented in RCR/C-PAL-01-02186. The licensee considered this issue a SCAQ because it met their procedure EN-LI-102 "Corrective Action Process" definition which stated the following definition for significant condition adverse to quality: "Conditions such as failures, malfunctions, deficiencies, deviations, defective material & equipment, and non-conformances which have resulted in, or could result in, a significant degradation or challenge to nuclear safety. The licensee concluded that the cracks in CRDM-21 were caused by TGSCC which occurred in areas of heavy grinding or machining tool marks. Specifically, this leak was the result of an inner diameter initiated, axially oriented, transgranular crack in the austenitic stainless steel housing material. The failure analysis performed in response to this event identified both axial and circumferential cracks associated with weld 3. Extent of condition inspections revealed additional, non-through wall cracks associated with weld 3 in 41 of the 44 remaining housings for a total of 42 of 45 housings containing cracks.

In response to the 2001cracking, Palisades replaced all 45 CRDM housings with housings thought to be more resistant to cracking. Principle changes included:

a. Elimination of weld number 2,

b. Relocation of weld number 3 to a higher location thereby minimizing the deposition of crud in the gap between the weld and the bottom plate of the rack and pinion assembly,

c. Reduction in residual stresses and cold work on welds by requiring better surface finishes, and

d. Use of heat sink welding to reduce ID residual tensile stresses.

Licensee corrective actions taken in response to the 2001 event were limited to pressure boundary welds and did not include the inspectors reviewed the licensee actions to determine if they had been sufficient to eliminate one of the 3 necessary factors to cause TGSCC on the CRDM housings: (1) a susceptible material, (2) a corrosive environment and (3) tensile stress." The inspectors identified that the licensee had failed to eliminate one or more of the necessary factors at the weld build-up area to preclude TGSCC in the replacement housing. Specifically:

- The licensee's 2001 root cause report documented that the weld build-up region is exposed to essentially the same environment as the weld that experienced the cracking (corrosive environment remained unchanged).

- No analysis was completed on the stress conditions for the weld build-up region prior to approving the modified replacement housing design (left residual tensile weld stresses on ID of CRDM surface).

- Fabrication restrictions to prohibit grinding were not applied to the weld build-up region (grinding promotes residual tensile stress state on ID of CRDM surface)

- Machining was performed on the weld build-up areas during the fabrication process in order to achieve the dimensions and geometry specified in the design. This process induced cold work stresses in the weld.

- Material was changed from type 347 to type 316 stainless steel (both materials are essentially equally susceptible to TGSCC).

Based upon the recurrence of through-wall leakage in the CRDM housings that occurred at the weld buildup region of the CRDM housings by TGSCC, the inspectors concluded that the licensee actions were not adequate because the appropriate actions to preclude recurrence were within the licensee's ability to foresee and implement. In 1991, the Fort Calhoun plant had experienced through-wall leakage due to TGSCC at the weld build-up

region of their CRDM housings (same housing design) and this operational experience had been reviewed by the licensee and dismissed. In the licensee's 2001 root cause evaluation, the licensee reviewed the weld build-up region failure by TGSCC at Fort Calhoun in the spare housing and concluded it would not occur at Palisades. This conclusion was based on the assumption that a higher oxygen environment (more aggressive environment) would exist in the spare Fort Calhoun housings than in the inservice Palisades housings. However the licensee did not confirm this assumption, nor did the licensee perform additional testing to determine if the environment of their inservice housings was sufficiently benign to prevent TGSCC. The licensee's 2012 RCT reached a similar conclusion and documented that due to organizational/ programmatic weakness at Palisades, the 1991 Fort Calhoun operating experience was not adequately utilized to include inspection of the housing ID weld build-up regions. The inspectors identified that the licensee had missed a key opportunity to implement effective corrective actions that could have prevented recurrence of the 2001 leakage event and elected not to pursue these actions because of the cost. Specifically, in EA-EAR-2001-0426-01 the licensee considered fabricating the replacement housings with Inconel 600 material because it was much more resistant to TGSCC. However, the licensee elected not to fabricate the replacement housings using this material because of the increased cost.

In January of 2002, an NRC special inspection team (SIT) (reference IR 50-2555/01-15) reviewed the licensee proposed corrective actions associated with the through-wall leakage of the CRDM-21 housing caused by TGSCC. The 2001 root cause report reviewed by the NRC stated the action to prevent recurrence was to "develop and implement an inspection plan to address areas and components identified in Attachment C-Extent of Condition". One of the components included in Attachment C was the CRD Mechanism. The recommended action was to perform volumetric inspection of the welds contained in the CRD Mechanism. The table also refers to a susceptibility analysis (EA-C-PAL-01-2186-02 "CRD Upper Housing and Nozzle Weld Susceptibility Comparison" to identify how degradation can be identified in this component. The objective of this document was to provide justification as to why the first weld (weld 1) above the reactor head is deemed to be less susceptible than the upper housing welds to failure by TGSCC and should not be included in the extent of condition. The susceptibility analysis excludes weld 5 because it is a weld overlay and not a butt weld and was deemed to be less susceptible to TGSCC than the butt welds. By not including weld 5 in the susceptibility analysis the licensee did not evaluate the stresses, material and environment of this weld to conclude it is not susceptible to TGSCC. An attachment to this analysis states machining marks were present on weld 5 which was identified as a key contributor to the cracking identified in weld 3. After this analysis was complete the licensee decided to replace all CRDM housings with the new design and control the fabrication process on the butt welds and the inspection plan would consist of the required ASME inspections. Weld 5 was excluded from these corrective actions and no fabrication controls were placed on weld 5 to reduce the stresses in this location. Therefore, the inspectors concluded that the licensee did not effectively implement corrective actions for the 2001 CRDM housing leak resulting in the 2012 CRDM-24 housing leak.

During the 2012 NRC special inspection the NRC identified an unresolved item for the Technical Specification pressure boundary leak. The licensee determined the CRDM-24 leakage commenced on July 14, 2012-and that the plant continued to operate in this condition for greater than 6 hours, which ~~is~~ was greater than the required shutdown time with pressure boundary leakage per TS LCO 3.4.14. Based on the review discussed above, unresolved items 05000255/2012012-01 "TS for PCS Pressure Boundary Leakage" and 05000255/2012012-03 "Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality," are closed.

Analysis: The inspectors determined that the licensee's failure to prevent recurrence of TGSCC of the CRDM housings (a significance condition adverse to quality) that resulted in a TS non-compliance was a performance deficiency that warranted a significance evaluation. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. Specifically the licensee did not limit the likelihood of events that upset plant stability by not taking adequate corrective actions to prevent recurrence of leakage in CRDM housings which represents a pressure boundary leakage and a condition prohibited by the Technical Specifications. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a ~~large~~ component rupture.

The inspectors determined this finding was caused by the same errors that led to the violation discussed in section 4OA2 (b.2) of this report and is indicative of current performance. Because the very similar cause for this performance deficiency and the one discussed in Section 4OA2 (b.2) of this report, no separate cross-cutting aspect is assigned to this finding.

Enforcement: The inspectors identified a NCV of 10 CFR Appendix B Criterion XVI "Corrective Actions", and Technical Specification 3.4.13 "Primary Coolant System Operational Leakage", having a very low safety significance (Green), for failure to prevent the recurrence of leakage in CRDM housings due to TGSCC resulting in the operation of the reactor with pressure boundary leakage for greater than the TS allowed time.

10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that, for significant conditions adverse to quality, the cause of the condition is determined and corrective action taken to preclude repetition.

Contrary to the above, August 12, 2012, the licensee failed to take corrective action to preclude repetition for a significant condition adverse to quality. Specifically, June 21, 2001 the licensee discovered a through wall leak in CRDM 21 due to TGSCC and failed to include weld 5 in the corrective actions as discussed in the above description which resulted in a through wall leak in CRDM 24. The pressure boundary leakage at CRDM began on July 14, 2012 and the plant continued to operate until August 12, 2012 which exceeded the 6 hours allowed by TS 3.4.13.

The licensee took corrective actions related to the results of the current root cause report which included the development of an inspection plan that would inspect weld 5 every outage until all CRDM housing were inspected.

Because this violation was of very low safety significance and it was entered into the licensee's corrective action program as CR-PLP-2013-01134, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000255/2013002-*xx*; Failure to Prevent Recurrence of a Significant Condition Adverse to Quality).

.2 Failure to Adequately Address the Generic Implications of the Cracking identified in CRDM 24

Introduction: The inspectors identified a Green Non-Cited Violation (NCV) of 10 CFR Part 50 Appendix B Criterion V, for failure to follow the root cause procedure. Specifically, the licensee failed to adequately evaluate the generic implications of the cause of the cracking identified in CRDM No. 24.

Description: While reviewing the 2012 root cause report CR-PLP-2013-05623 related to the cracking identified in CRDM No. 24 the inspectors identified that the licensee had not appropriately considered the generic implications of the cracking in the extent of condition review. The licensee's proposed corrective actions narrowly focused on the weld build up region (weld 5), instead of broader actions to ensure other CRDM housing welds were fit for their intended service life.

On March 13, the inspectors requested that the licensee identify the bases for excluding other CRDM housing welds (weld #3 below the weld build up region and weld #4 above the weld build up region) from the scope of planned corrective actions. On March 29, the licensee provided additional information to justify excluding these welds from the scope of the corrective actions. The licensee credited the corrective actions associated with the modifications to the CRDM housing design completed in 2001 as the basis to exclude housing welds No 3 and 4 from additional actions to identify the extent of TGSCC. The corrective actions taken in 2001 included performing heat sink welding, which is a methodology used to reduce the stresses on the inner diameter (ID) of the weld, they also changed the design to reduce design stresses at weld #3 and they specified a smoother surface finish (RMS 125) to reduce potential crack initiation points. The licensee stated that these actions would produce compressive stresses on the ID of welds 3 and 4 making them immune from cracking. The inspectors acknowledged that these actions would reduce the tensile stress at the ID surface and thus reduce the probability of initiating TGSCC.

However, the information provided did not demonstrate that TGSCC would not occur because it did not demonstrate that tensile stress would be eliminated at the ID surface during operation. In particular, repairs completed at the inner surface of weld No. 4, would result in high residual tensile stress at the inside surface of the weld which would promote the initiation of TGSCC. Repairs were also performed on weld No. 3; from the outer diameter (OD) surface of the weld. The licensee believed that the last pass heat sink welding process would be sufficient to ensure residual compressive stress would remain at the ID surface of Weld No. 3 even with repairs to the OD surface. However, the licensee had not completed detailed residual weld stress testing or modeling to confirm this assumption.

The inspectors identified that the three factors required for TGSCC could still be present at the welds 3 and 4 as follows:

- Corrosive environment – Weld 3 would operate in a similar environment as the weld build up region of the CRDM housing. Weld No. 4 would be exposed to a lower operating temperature then the weld build up region, however TGSCC can still occur at 250 degrees Fahrenheit as evidenced by the Palisades previous operating experience with cracking identified in the seal housings that operate at even lower temperatures.

- Susceptible material – Welds 3, 4 and 5 are composed of the same weld filler and base metal materials as the weld buildup region (e.g. weld filler material consistent with the type 316 stainless housing base metal). This material would be equally susceptible to TGSCC, as the type 347 stainless steel and weld filler materials used in the pre-2001 CRDM housing design that developed a through wall leak caused by TGSCC at weld No.3.

- Tensile stresses - While it is assumed that the corrective actions taken in response to the 2001 leak will reduce the potential for tensile stresses to exist on the inner surface of CRDM housings at welds 3 and 4, especially in light of the repairs made to welds 3 and 4, it has not been conclusively demonstrated that these tensile stresses have been eliminated. As such it is not reasonable to conclude that tensile stresses are not present and, therefore, the potential for transgranular stress corrosion cracking has been eliminated.

Although the root cause report discusses manufacturing irregularities and misalignment between CRDM housing 24 and the support tube, seismic supports and the associated reactor head penetration/CRDM nozzle as potential source of stresses leading to cracking, the root cause report also states that "based on the lack of cracking found in the other 8 upper housings tested, the failed CRD-24 upper housing contains an as-yet unidentified additional stress." Because the cause of the additional stress was not identified, the licensee had not established a sufficient basis in the RCR to exclude welds 3 and 4 from the extent of condition review (e.g. potential generic implications).

The inspectors identified that the licensee had not followed Procedure EN-LI-118 Root Cause evaluation in the root cause review of the CRDM housing No. 24 leak as documented in report CR-PLP-2013-05623. Section 5.5 (12)e of EN-LI-118 required that the licensee "perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affects other SSC's." In this case, the inspectors identified that the licensee had not documented a sufficient basis in RCR CR-PLP-2013-05623 to exclude welds No. 3 and No. 4 from the generic factors discussed above that led to the 2012 leak in the CRDM housing No. 24 (e.g. TGSCC at the weld buildup region). The licensee entered this issue into the corrective action program as CR-PLP-2013-01500. To restore compliance with the procedure, the licensee intended to revise the inspection plan to add additional corrective actions to inspect a sample of welds No. 3 and 4 for TGSCC during the upcoming refueling outage.

Analysis: The inspectors determined that the failure to adequately evaluate the generic implications of the cause of the cracking identified in CRDM #24 in accordance with the root cause procedure EN-LI-118 was a performance deficiency that warranted a significance evaluation. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. The inspectors also answered "yes" to the More-than-Minor screening questions "if left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, absent NRC identification, the licensee would not have completed further evaluations or inspections of CRDM housing welds which could have resulted in additional CRDM housing failure and leakage by TGSCC. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a large component rupture.

The inspectors determined that the primary cause of the failure to adequately consider welds 3 and 4 on the generic implications section of the root cause report related to the cross-cutting component of Human Performance, Decision Making, because licensee staff did not use conservative assumptions in decision making. Specifically, the licensee did not use conservative assumptions when excluding welds 3 and 4 as being susceptible to TGSCC and therefore include them in the generic implications section of the root cause report. (Item H.1(b) of IMC 310).

Enforcement: The inspectors identified a NCV of 10 CFR Appendix B Criterion V "Instructions, Procedures and Drawings", having a very low safety significance (Green), for failure to adequately evaluate the generic implications of the cause of cracking identified in CRDM #24 as it relates to weld 3 and 4 in accordance with the root cause procedure.

Title 10 CFR Appendix B Criterion V "Instruction, Procedures and Drawings requires in part, "Activities affecting quality shall be prescribed by documented procedures and shall be accomplished in accordance with these procedures."

Procedure EN-LI-118 Root Cause evaluation process revision 17 states:

a. 5.5 (12)e: perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affects other SSC's , organizations or work processes. Use the two step process in accordance with attachment 9.7
b. Attachment 9.7 states Determine whether the occurrence/consequence (problem) is isolated, or whether it has broader (generic or common mode) implications. Achieve this by asking the following questions:
    i. Could this happen to equipment that is similar in function, design, or service condition?
    ii. Could this happen to a group of components? (components of the same construction or materials that could be similarly affected by one condition)
c. Attachment 9.7 also states: Document the results of the above considerations. Include the following items in the write up:
    i. Generic Implications (Is this problem/ cause limited to this component/equipment, or does it apply to others as well)
    ii. Existing broader (generic/common mode) considerations
d. 5.5(15)(10)c&f: Document proposed corrective actions and due dates to address valid generic implications. If no corrective action is recommended for a valid generic implication then document the basis for this conclusion and any risk or consequence identified as a result of taking no action.

Contrary to the above, from February 24, 2013 through April 18, 2013, the licensee failed to perform an activity affecting quality in accordance with procedure EN-LI-118. Specifically, the licensee did not identify and document the existing broader (generic/common mode) considerations associated with TGSCC at CRDM housing welds No. 3 and No. 4. Consequently, the licensee failed to propose corrective actions for the generic implications of TGSCC at CRDM housing welds No. 3 and No. 4. The licensee was considering adding welds 3 and 4 into their inspection plan for activities to be performed during the next refueling outage. Because of the very low safety significance and because the licensee entered this issue into their corrective action program (CR-PLP-2013-01500), it is being treated as a NCV consistent with Section 2.3.2 of the Enforcement Policy (NCV 05000255/2013003-xx).

4OA5 Other Activities

.1 (Closed) Unresolved Item 05000255/2012012-02: Potential Inadequate Degradation Evaluation of CRDM Housings

During a Special Inspection performed in August 2012, NRC inspectors identified an issue which could not be resolved without additional information (Unresolved Issue (URI)). This issue was associated with the rate of growth of the crack which created the through wall leak in CRDM housing 24, discovered on August 12, 2012. Identification of this crack growth rate is significant in determining appropriate intervals for future inspections to provide reasonable assurance that CRDM housing leakage will not recur.

Preliminary failure analysis data available at the time of the inspection indicated that the observed cracking was due to transgranular stress corrosion cracking. Cracking of this type is normally due to the presence of oxygen and chlorides at the location of the crack. When examining the fracture surface at the location the through-wall leak occurred, the licensee identified six concentric rings (beach marks) propagating in a radial direction from the inside diameter out towards the outside diameter of the housing. Beach marks are normally associated with fatigue failures and indicate the number of stress cycles from crack initiation to crack failure. In this case there was no evidence that fatigue contributed to the failure. Despite the lack of evidence of fatigue, it was apparent that the crack which resulted in the CRDM housing 24 leak grew in increments. It was not, however, immediately apparent whether the increments were related to oxygen ingress (refueling outages) or temperature/pressure cycles.

At the time of the original inspection, 5 time intervals for through wall crack growth were under consideration. Two were based on literature crack growth data and three were based on interpretations of the beach marks. These time intervals were:

1. Based on literature data, one contractor estimated that a 10% through wall flaw would require 4 years to reach 50% through wall.
2. Based on literature data another contractor estimated the crack growth rate to be 2.1 x 10^{-5} in/hr or 0.18 in/yr. This is approximately three times faster than the crack growth rate proposed in the above mentioned rate.
3. Based on the concept of oxygen ingress at refueling outages 6 cycles of 18 months duration would require 9 years for the crack to grow through wall
4. Based on the concept of temperature/pressure cycles, the plant experienced 6 cold shutdowns in approximately 2 years preceding the crack. This equates to 2 years for the crack to grow through wall.
5. Based on the concept that oxygen is required for crack growth and that oxygen is rapidly purged from the CRDM housings due to leakage past the seals, crack growth occurs only during the first few weeks of operation following a refueling outage, followed by no growth for the remaining period of operation when oxygen concentrations are low. This equates to 6 oxygen ingress events (irrespective of time between events) for the crack to grow through wall.

NRC inspectors including technical experts from NRC Headquarters performed a follow-up inspection to determine if the assumptions made by the licensee were conservative and the planned actions bounded those conservative assumptions. The inspectors reviewed a variety of documents associated with crack growth and inspection intervals.

The inspectors noted various the following statements included in the root cause report and vendor documents related to the determination of the appropriate crack growth rate.

1. The laboratory conducting the failure analysis concluded, it could not be conclusively determined if the beach marks corresponded to refueling outages, (i.e., 18 month cycle) or shorter periods as occurred during outages over the past 24 months
2. Palisades CRDM housing 21 leaked at weld 3 in 2001. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 failure. In calculating the crack growth rate of this crack, one contractor utilized an interval between beach marks which is much shorter than refueling outages. The intervals used are consistent with plant thermal cycles in which oxygen may or may not have been admitted into the CRDMs.
3. A spare CRDM housing at Ft Calhoun leaked at weld 5 in 1990. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 failure. In calculating the crack growth rate of this crack, Ft Calhoun stated that the beach marks were related to refueling cycles. Ft Calhoun also performed calculations indicating that the oxygen level at the location of the flaw did not change with time (including in response to refueling outages) because the spare CRDM housing was not vented. Ft Calhoun's evaluation indicated that oxygen levels at the vicinity of the crack would have begun to decline through diffusion and convection had the intervals between outages been much longer than 18 months. This is interpreted to mean that the beach marks at Ft Calhoun are in response to pressure/thermal cycles.
4. In at least one instance Palisades needed to repair the seals on a reactor coolant pump at a time other than an outage. This necessitates draining some of the water from the reactor coolant system and venting (admitting oxygen into) the CRDM housing. This represents an additional oxygen ingress event not included when determination of time to cracking is based on refueling outages.
5. In its inspection plan Palisades states that it will inspect all CRDM housings over the next 4 refueling outages, i.e., the interval between inspections is 1 refueling outage

Based on the above review, the inspection team notes that there are certain non conservative statements contained in the Root Cause Report and the inspection plan. These include:

1. The crack growth rate based on refueling outages is understated. If oxygen ingress is related to beach marks, given the oxygen ingress event which occurred to repair reactor coolant pump seals, 6 beach marks would occur in a maximum of 5 refueling intervals rather than the 6 refueling intervals that were used to calculate the crack growth rate in the root cause report.
2. The crack growth rate based on heat up and cool down cycles is overstated. The value in the root cause is based on 11 months. While 6 shutdowns did occur at the plant in 11 months several of these events did not result in pressure/temperature changes of the reactor coolant system. The appropriate time frame is 24 months rather than 11.
3. The inspection plan contains a non conservative statement: "However, once the crack has been initiated it propagates over 4 to 5 operating cycles prior to going through wall." While this statement does reflect one of the proposed theories for crack growth, sufficient evidence to demonstrate reasonable assurance that this

theory is correct, and thereby overcome the non-conservatism of this statement, does not exist.

Despite the existence of the non conservatisms stated above, the inspectors conclude:

1. Sufficient evidence to conclusively determine the rate of crack growth does, and will not exist.
2. Crack growth based on pressure/temperature cycles is the most conservative of the potential crack growth mechanisms. In the absence of reasonable assurance of the correctness of less conservative mechanisms, through wall crack growth in 2 years must be utilized for regulatory purposes.
3. The licensee has not formally committed to any of the crack growth mechanisms discussed.
4. The licensee's inspection program includes inspections in each of the next 4 outages. This inspection interval, once per outage, bounds all the crack growth mechanisms considered.

The staff finds this approach to inspection to be both acceptable and sufficient justification to close this URI.

4OA6 Management Meetings

.2 Interim Exit Meetings

An interim exit was conducted for:

- The results of the selected issue follow-up inspection, with Mr. T. Vitali, Site Vice President on April 18, 2013.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

B. Davis, Engineering Director
O. Gustafson, Licensing Manager
T. Foudy, Engineering Supervisor
B. Williams, Engineer
B. Dotson, Licensing

LIST OF ITEMS OPENED, CLOSED, DISCUSSED

Closed

05000255/2012012-01	URI	TS for PCS Pressure Boundary Leakage
05000255/2012012-02	URI	Potential Inadequate Degradation Evaluation of CRDM Housings
05000255/2012012-03	URI	Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality

Opened and Discussed

None.

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

4OA5 Other Activities

LIST OF ACRONYMS USED

Giessner, John RELEASE ENTIRETY

From: Sanchez Santiago, Elba
Sent: Tuesday, April 23, 2013 8:08 AM
To: Taylor, Thomas; Giessner, John
Subject: RE: Call on CRD yesterday

For the Criterion XVI issue EICS and Pat Louden are in agreement with the characterization of the issue.

For the Criterion V issue EICS is pushing back on whether we can give a violation against the root cause procedure. They will be reaching out to their counterparts in headquarters to see what their take on the issue is. Meanwhile we are exploring other options for characterizing this issue. More to come.

-Elba

From: Taylor, Thomas
Sent: Tuesday, April 23, 2013 7:30 AM
To: Giessner, John; Sanchez Santiago, Elba
Subject: Call on CRD yesterday

How did it go?

tom

Tom Taylor
US NRC Senior Resident Inspector
Palisades Nuclear Plant
269-764-8971 (w)
Thomas.Taylor@nrc.gov

C3

Release entirety

Giessner, John

From: Sanchez Santiago, Elba
Sent: Monday, April 29, 2013 11:31 AM
To: Hills, David; Orth, Steven; Giessner, John
Subject: Palisades input (CRDM inspection)
Attachments: Palisades Input to DRP Report 2013 002 URI EMS.docx

Follow Up Flag: Follow up
Flag Status: Completed

All,

Attached is the latest revision to the input to the Palisades quarterly report. I included the changes recommended as a result of our discussions. Please review and let me know if you have any comments or questions.

Thanks,

Elba M. Sanchez Santiago
Reactor Engineer
RIII/ DRS/ EB1
630-829-9715

C32

UNITED STATES
NUCLEAR REGULATORY COMMISSION
LISLE, IL 60532-4352

April XX, 2012

MEMORANDUM TO: Thomas Taylor
Senior Resident Inspector
Palisades Nuclear Plant

FROM: David Hills, Chief
Engineering Branch 3
Division of Reactor Safety

SUBJECT: PALISADES NUCLEAR PLANT
DRS INPUT TO INTEGRATED REPORT 05000255/2013002

Enclosed is the report input for the Palisades Nuclear Plant, Inspection Report 05000255/2013002. This report input documents completion of our review of Unresolved Items 05000255/2012012-01, "TS for PCS Pressure Boundary Leakage," 05000255/2012012-02, "Potential Inadequate Degradation Evaluation of CRDM Housings," and 05000255/2012012-03, "Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality." This report also completes one sample of the Problem Identification and Resolution, Selected Issue Follow-up inspection in accordance with IP 71152. I have reviewed this input to confirm compliance with Inspection Manual Chapter (IMC) 0612 and IMC 0305. This input is ready for inclusion into the integrated report and dissemination to the public.

Please input the following post Inspection Data into RPS:

Inspection Procedure	**Procedure Status** – see below: Incomplete, Complete, Complete by reference, Complete-full sample not available, Complete – opportunity to apply procedure not available, Not Applicable.	**Sample Size** – As documented in Scope Section If less than full sample size documented in the report input, the inspector must provide a justification below to enter into RPS and support the procedure status selected
71152	Complete	1

Inspection Report Item and Type (AV, FIN, NCV, URI or VIO)	**Cornerstone** (IE, MS, BI, EP, OR, PR, MISC)	**Cross Cutting Aspect** (H.n(i), P.n(i), S.n(i))	**Responsible Person/Owner**	**Procedure or TI** (71111.07T)	**RPS Branch Code (e.g. closeout responsibility)** EB1 3820 EB2 3870 EB3 3840 PST (RP) 3860 PSB (Safeguards) 3850 OB 3810
NCV-XXX	IE	n/a	E. Sanchez Santiago	71152	3820
NCV-XXX	IE	H.1(b)	E. Sanchez Santiago	71152	3820

Enclosure: Input to Inspection Report 05000255/2013002

cc w/encl: J. Giessner, Chief
C. Hernandez, Site Admin Assistant

CONTACT: E. Sanchez Santiago, DRS
(630) 829-9715

DOCUMENT NAME: G:\DRSIII\DRS\Work in Progress\-Palisades Input to DRP Report 2013 002 URI EMS.docx
9 Publicly Available 9 Non-Publicly Available 9 Sensitive 9 Non-Sensitive
To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	RIII		NRR		RIII		NRR	
NAME	ESanchezSantiago		DAlley		DHills		TLupold	
DATE	4/ /13							

OFFICIAL RECORD COPY

Cover Letter

X Green findings involving a violation were identified. Include the following:

Based on the results of this inspection, two NRC-identified findings of very low safety significance (Green) were identified. These findings were determined to involve a violation of NRC requirements. However, because of the very low safety significance and because the issues were entered into your corrective action program, the NRC is treating the issue as Non-Cited Violation, in accordance with Section 2.3.2 of the NRC Enforcement Policy.

TITLE PAGE

Inspectors: D. Alley, Senior Materials Engineer
E. Sanchez Santiago, Reactor Inspector

SUMMARY OF FINDINGS

A. NRC-Identified and Self-Revealed Findings

Cornerstones: Initiating Events

- Green. The inspectors identified a Finding with associated Non-Cited Violations (NCV) of 10 CFR Part 50, Appendix B, Criterion XVI and Technical Specification (TS) 3.4.13 Primary Coolant System (PCS) Operational Leakage for failure to prevent recurrence of CRDM cracking and leakage, a significant condition adverse to quality which resulted in a violation of TS. Specifically, the licensee failed to include the internal CRDM housing weld build-up area within the scope of corrective actions taken for a 2001 CRDM leakage event and consequently leakage recurred in CRDM-24 in 2012.

  The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. Specifically the licensee did not limit the likelihood of events that upset plant stability by not taking adequate corrective actions to prevent recurrence of leakage in CRDM housings which represents a pressure boundary leakage. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System LOCA initiator contributor. The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a component rupture. Despite the advanced age of the licensees decisions associated with this finding, the inspectors concluded that the finding was indicative of current performance. Specifically, the licensee more recently exhibited similar non-conservative decision making with respect to addressing the potential for CRDM housing cracking and leakage (Section 4OA2.3 (b.2) of this report) and resulting in another finding. However, given that both findings reflect upon the licensee's approach to basically the same equipment and technical issues, the inspectors did not apply a separate cross cutting aspect to this finding in that it is already captured through the other finding. (Section 4OA2.3(b.1))

- Green. The inspectors identified a Finding with an associated Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion V, for failure to follow the root cause procedure. Specifically, the licensee failed to adequately evaluate and document the generic implications of the cause of the cracking identified in CRDM-24.

  The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. The inspectors also answered "yes" to the More-than-Minor screening question "if left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, absent NRC identification, the licensee would not have completed further evaluations or inspections of CRDM housing welds which could have resulted in additional CRDM housing failure and leakage by TGSCC. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System LOCA initiator contributor. The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609, Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a large component rupture. The inspectors determined that the primary cause of the failure to adequately consider welds No. 3 and No. 4 in the generic implications section of the root cause report related to the cross-cutting component of Human Performance, Decision Making, because licensee staff did not use conservative assumptions in decision making. Specifically, the licensee did not use conservative assumptions when excluding welds No. 3 and No. 4 as being susceptible to TGSCC and therefore include them in the generic implications section of the root cause report. (Item H.1(b)). (Section 4OA2.3(b.2))

B. Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAILS

4. REACTOR SAFETY

4OA2 Identification and Resolution of Problems (71152)

.3 Selected Issue Follow-up Inspection: Through Wall Leakage of Control Rod Drive Mechanism (CRDM) Housing #24

a. Inspection Scope

On August 12, 2012, the licensee shut down the plant to investigate an increase in unidentified leakage. The source of the leakage was determined to be a crack in CRDM-24. The NRC dispatched a special inspection team (SIT) to review the CRDM-24 leakage event. The results of that inspection are provided in Inspection Report 05000255/2012012. The licensee completed an evaluation to determine the cause of the cracking (CR-PLP-2012-05623).

From March 4, 2013 to March 15, 2013, the inspectors completed one inspection sample regarding problem identification and resolution based upon review of the licensee's root cause report contained in corrective action document CR-PLP-2012-05623. In addition the inspectors performed reviews related to three Unresolved Items (URI) identified during the SIT inspection:

- URI 05000255/2012012-01 TS for PCS Pressure Boundary Leakage. (The closure of this URI is documented in section 4OA2.3 (b.1) of this report.)
- URI 05000255/2012012-02 Potential Inadequate Degradation Evaluation of CRDM Housings (The closure of this URI is documented in section 4OA5.1 of this report)
- URI 05000255/2012012-03 Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality (The closure of this URI is documented in section 4OA2.3 (b.1) of this report.)

The inspectors reviewed the licensee's actions in accordance with performance attributes identified in IP 71152. Specifically, the inspectors reviewed licensee corrective action records to determine if: (1) the problems were accurately identified; (2) operability and reportability were adequately ascertained; (3) extent of condition and generic implications were appropriately addressed; (4) classification and prioritization of the problem were commensurate with safety significance; (5) root and contributing causes were identified; (6) corrective actions were appropriately focused to correct the problem; and (7) timely corrective actions were completed or proposed commensurate with the safety significance of the issues.

b. Findings

.1 Failure to Prevent Recurrence of CRDM Housing Cracking and Leakage

Introduction: The inspectors identified a Green Finding with associated Non-Cited Violations (NCV) of 10 CFR Part 50, Appendix B, Criterion XVI and Technical

Specification (TS) 3.4.13 Primary Coolant System (PCS) Operational Leakage for failure to prevent recurrence of CRDM cracking and leakage, a significant condition adverse to quality which resulted in a violation of TS. Specifically, the licensee failed to include the internal CRDM housing weld build-up area within the scope of corrective actions taken for a 2001 CRDM leakage event and consequently leakage recurred in CRDM-24 in 2012.

Description: In 2001, the licensee discovered a steam leak in the housing of CRDM-21 caused by a through-wall TGSCC at CRDM housing weld No. 3 which was located just below the weld build-up region (weld No. 5). Weld No. 5 consists of a weld material deposit applied to the inside diameter (ID) of the CRDM housing which provides for alignment of the CRDM. This issue was categorized as a significant condition adverse to quality (SCAQ) by the licensee (CPAL0102186) and the licensee's root cause evaluation was documented in RCR/C-PAL-01-02186. The licensee considered this issue a SCAQ based on the procedure EN-LI-102 "Corrective Action Process" definition of "Conditions such as failures, malfunctions, deficiencies, deviations, defective material & equipment, and non-conformances which have resulted in, or could result in, a significant degradation or challenge to nuclear safety." The licensee concluded that the cracks in CRDM-21 were caused by TGSCC which occurred in areas of heavy grinding or machining tool marks. Specifically, this leak was the result of an ID initiated, axially oriented, transgranular crack in the austenitic stainless steel housing material. The failure analysis performed in response to this event identified both axial and circumferential cracks associated with weld No. 3. Extent of condition inspections revealed additional, non-through wall cracks associated with weld No. 3 in 41 of the 44 remaining housings for a total of 42 of 45 housings containing cracks.

In response to the 2001cracking, Palisades replaced all 45 CRDM housings with housings thought to be more resistant to cracking. Principle changes included:

- Elimination of weld No. 2,
- Relocation of weld No. 3 to a higher location thereby minimizing the deposition of crud in the gap between the weld and the bottom plate of the rack and pinion assembly,
- Reduction in residual stresses and cold work on welds by requiring better surface finishes, and
- Use of heat sink welding to reduce ID residual tensile stresses.

Licensee corrective actions taken in response to the 2001 event were limited to butt welds. The inspectors reviewed the licensee actions to determine if they had been sufficient to eliminate one of the 3 necessary factors to cause TGSCC on the CRDM housings: (1) a susceptible material, (2) a corrosive environment and (3) tensile stress. The inspectors identified that the licensee had failed to eliminate one or more of the necessary factors at weld No. 5 (which was not a butt weld) to preclude TGSCC in the replacement housing. Specifically:

- The licensee's 2001 root cause report documented that weld No. 5 is exposed to essentially the same environment as the weld that experienced the cracking (corrosive environment remained unchanged).

- No analysis was completed on the stress conditions for weld No. 5 prior to approving the modified replacement housing design (the potential for residual tensile weld stresses on ID of CRDM surface was not ruled out by analysis and therefore, should have been considered).

- Fabrication restrictions to prohibit grinding were not applied to weld No. 5 (grinding promotes residual tensile stress state on ID of CRDM surface)

- Machining was performed on weld No. 5 during the fabrication process in order to achieve the dimensions and geometry specified in the design. This process induced cold work stresses in the weld.

- Material was changed from type 347 to type 316 stainless steel (both materials are essentially equally susceptible to TGSCC).

In January of 2002, an NRC special inspection team (SIT) (reference IR 50-2555/01-15) reviewed the licensee proposed corrective actions associated with the through-wall leakage of the CRDM-21 housing caused by TGSCC. The 2001 root cause report reviewed by the NRC stated the action to prevent recurrence was to "develop and implement an inspection plan to address areas and components identified in Attachment C-Extent of Condition". One of the components included in Attachment C was the CRD Mechanism. The recommended action was to perform volumetric inspection of the welds contained in the CRD Mechanism. The table also refers to a susceptibility analysis (EA-C-PAL-01-2186-02 "CRD Upper Housing and Nozzle Weld Susceptibility Comparison") to identify how degradation can be identified in this component. The objective of this document was to provide justification as to why the first weld (weld No. 1) above the reactor head is deemed to be less susceptible than the upper housing welds to failure by TGSCC and should not be included in the extent of condition. The susceptibility analysis excludes weld No. 5 because it is a weld overlay and not a butt weld and was deemed to be less susceptible to TGSCC than the butt welds. By not including weld No. 5 in the susceptibility analysis the licensee did not evaluate the stresses, material and environment of this weld to conclude it is not susceptible to TGSCC. An attachment to this analysis states machining marks were present on weld No. 5 which was identified as a key contributor to the cracking identified in weld No. 3. After this analysis was complete the licensee decided to replace all CRDM housings with the new design and control the fabrication process on the butt welds and the inspection plan would consist of the required ASME inspections. Weld No. 5 was excluded from these corrective actions and no fabrication controls were placed on weld No. 5 to reduce the stresses in this location.

On August 12, 2012, Palisades Nuclear Power Station was shut down to investigate an increase in unidentified leakage. During a walk-down performed post shutdown, the licensee discovered the source of the leakage to be a pressure boundary leak from CRDM-24. After further testing, the licensee determined the leak occurred because of a through-wall crack adjacent to weld No. 5.

The licensee formed a root cause team (RCT) staffed with licensee personnel and augmented with input from vendors. The root cause investigation was conducted in accordance with site procedure EN-LI-118 "Root Cause Evaluation Process" and was documented in root cause analysis report CR-PLP-2012-05623. In this report, the licensee's RCT determined that the probable cause of the cracking was:

> "Stresses in the weld build up area due to manufacturing irregularities and misalignments between CRDM-24 upper housing, support tube, and the associated reactor head penetration/CRDM nozzle. Based on lack of cracking found in the other 8 upper housings tested, the failed CRDM-24 upper housing contains an as-yet unidentified additional stress."

The RCT also identified the following contributing cause:

> "Transgranular Stress Corrosion Cracking (TGSCC) initiating within the internal weld build-up material of CRDM-24. The through wall crack initiated in the weld material and then propagated through the base metal until a leak developed in the outer diameter (OD) witness band region at the base of the ID weld build up.

This conclusion was based upon destructive and non destructive examinations (NDE) completed on a section of the failed housing which included the through-wall flaw. The RCT also relied upon vendor technical reports assessing the results of the NDE as well as vendor calculations related to the stresses in the CRDM housings.

To determine the extent of condition, the licensee performed ultrasonic (UT) examinations of weld No. 5 on eight additional CRDM housings. The licensee selected these housings based on being in a similar location on the head as CRDM-24, and previous cracking having been identified in some of these housings prior to the replacement of the CRDM upper housings and seal housings in 2002. The inspectors concluded that this was an adequate sample for an initial extent of condition review based upon the concept that, in light of eight negative exams, the statistical probability of a flaw in the remaining CRDM housings was very low. Additionally, the licensee planned to conduct examinations of more housings during the next refueling outage.

Based upon the recurrence of through-wall leakage in the CRDM housings caused by TGSCC, the inspectors concluded that the licensee actions were not adequate because the appropriate actions to preclude recurrence were within the licensee's ability to foresee and implement. Specifically, the inspectors concluded that the licensee did not effectively implement corrective actions for the 2001 CRDM housing leak resulting in the 2012 CRDM-24 housing leak. Also, in 1991, the Fort Calhoun plant had experienced through-wall leakage due to TGSCC at weld No. 5 of their CRDM housings (same housing design) and this operational experience had been reviewed by the licensee and dismissed. In the licensee's 2001 root cause evaluation, the licensee reviewed the weld build-up region failure by TGSCC at Fort Calhoun in the spare housing and concluded it would not occur at Palisades. This conclusion was based on the assumption that a higher oxygen environment (more aggressive environment) would exist in the spare Fort Calhoun housings than in the inservice Palisades housings. However the licensee did not confirm this assumption, nor did the licensee perform additional testing to determine if the environment of their inservice housings was sufficiently benign to prevent TGSCC. The licensee's 2012 RCT reached a similar conclusion and documented that due to organizational/ programmatic weakness at Palisades, the 1991 Fort Calhoun operating experience was not adequately utilized to include inspection of the weld No. 5. The inspectors identified that the licensee had missed a key opportunity to implement effective corrective actions that could have prevented recurrence of the 2001 leakage event and elected not to pursue. Specifically, in EA-EAR-2001-0426-01 the licensee considered fabricating the replacement housings with Inconel 600 material because it was much more resistant to TGSCC, but ultimately decided not to do so. Additionally, various vendor reports were generated related to this issue. Those reports documented the potential susceptibility of weld No. 5 to TGSCC due to their review of the CRDM housing conditions and available operating experience. The issuance of these documents represents another opportunity for the licensee to identify the susceptibility of weld No. 5 to TGSCC prior to the cracking in CRDM-24.

During the 2012 NRC special inspection, the NRC identified an unresolved item for the Technical Specification pressure boundary leak. The licensee determined the CRDM-24 leakage commenced on July 14, 2012 and the plant continued to operate in this condition, which was contrary to the TS 3.4.13 requirement of limiting PCS operational leakage to no pressure boundary leakage. Based on the review discussed above, unresolved items 05000255/2012012-01 "TS for PCS Pressure Boundary Leakage" and 05000255/2012012-03 "Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality" are closed.

Analysis: The inspectors determined that the licensee's failure to prevent recurrence of TGSCC of the CRDM housings (a significant condition adverse to quality) that resulted in a violation of TS was a performance deficiency that warranted a significance evaluation. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. Specifically the licensee did not limit the likelihood of events that upset plant stability by not taking adequate corrective actions to prevent recurrence of leakage in CRDM housings which represents a pressure boundary leakage. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a component rupture.

Despite the advanced age of the licensees decisions associated with this finding, the inspectors concluded that the finding was indicative of current performance. Specifically, the licensee more recently exhibited similar non-conservative decision making with respect to addressing the potential for CRDM housing cracking and leakage (Section 4OA2.3 (b.2) of this report) and resulting in another finding. However, given that both findings reflect upon the licensee's approach to basically the same equipment and technical issues, the inspectors did not apply a separate cross cutting aspect to this finding in that it is already captured through the other finding.

Enforcement: The inspectors identified NCVs of 10 CFR, Appendix B, Criterion XVI "Corrective Actions", and Technical Specification 3.4.13 "Primary Coolant System Operational Leakage", having a very low safety significance (Green), for failure to prevent the recurrence of leakage in CRDM housings due to TGSCC resulting in the operation of the reactor with pressure boundary leakage, a condition prohibited by TS. Given that both violations relate to the same performance deficiency, they are considered as one finding.

10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that, for significant conditions adverse to quality, the cause of the condition is determined and corrective action taken to preclude repetition.

Technical Specifications 3.4.13 requires PCS operational leakage be limited to "No pressure boundary LEAKAGE" when in Modes 1 through 4.

Contrary to the above, as of August 12, 2012, the licensee had failed to take corrective actions to preclude repetition for a significant condition adverse to quality. Specifically, June 21, 2001, the licensee discovered a through wall leak in CRDM-21 due to TGSCC and failed to reasonably include weld No. 5 in the corrective actions which resulted in a subsequent through wall leak in CRDM-24. The pressure boundary leakage at CRDM-24 began on July 14, 2012, and the plant continued to operate until August 12, 2012, which is contrary to the TS requirement of limiting operational leakage to no pressure boundary leakage.

As a result of the second through wall leak, the licensee took corrective actions which included the development of an inspection plan that would inspect weld No. 5 every outage until all CRDM housings were inspected.

Because these violations were of very low safety significance and were entered into the licensee's corrective action program as CR-PLP-2013-01134, these violations are being treated as an NCVs, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000255/2013002-*xx*; Failure to Prevent Recurrence of a Significant Condition Adverse to Quality).

.2 Failure to Adequately Address the Generic Implications of the Cracking identified in CRDM 24

Introduction: The inspectors identified a Green Finding with an associated Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion V, for failure to follow the root cause procedure. Specifically, the licensee failed to adequately evaluate and document the generic implications of the cause of the cracking identified in CRDM-24.

Description: As a result of the cracking identified in CRDM-24, which was characterized as a SCAQ, the licensee performed a root cause evaluation in accordance with procedure EN-LI-118, "Root Cause Evaluation". This procedure is identified as quality related and serves to implement a portion of the licensee's quality assurance program. While reviewing the 2012 root cause report CR-PLP-2013-05623 related to the cracking identified in CRDM-24, generated as a result of the root cause evaluation, the inspectors identified that the licensee had not appropriately considered the generic implications of the cracking in the extent of condition review. The licensee's proposed corrective actions narrowly focused on weld No. 5, instead of also including broader actions to ensure other CRDM housing welds were fit for their intended service life.

On March 13, 2013, the inspectors requested that the licensee provide the bases for excluding other CRDM housing welds (weld No. 3 below weld No. 5 and weld No. 4 above weld No. 5) from the scope of planned corrective actions. On March 29, the licensee provided additional information to justify excluding these welds from the scope of the corrective actions. The licensee credited the corrective actions associated with the modifications to the CRDM housing design completed in 2001 as the basis to exclude housing welds No. 3 and No. 4 from additional actions to identify the extent of TGSCC. The corrective actions taken in 2001 included performing heat sink welding, which is a methodology used to reduce the stresses on the inner ID of the weld. The licensee also changed the design to reduce design stresses at weld No. 3 and specified a smoother surface finish (RMS 125) to reduce potential crack initiation points. The licensee stated that these actions would produce compressive stresses on the ID of welds No. 3 and No. 4 making them immune from cracking. The inspectors acknowledged that these actions would reduce the tensile stress at the ID surface and thus reduce the probability of initiating TGSCC.

However, the information provided did not demonstrate that TGSCC would not occur because it did not demonstrate that tensile stress would be eliminated at the ID surface during operation. In particular, repairs completed at the inner surface of weld No. 4, would result in high residual tensile stress at the inside surface of the weld which would promote the initiation of TGSCC. Repairs were also performed on weld No. 3 from the OD surface of the weld. The licensee believed that the last pass heat sink welding process would be sufficient to ensure residual compressive stress would remain at the ID surface of Weld No. 3 even with repairs to the OD surface. However, the licensee had not completed detailed residual weld stress testing or modeling to confirm this assumption.

The inspectors identified that the three factors required for TGSCC could still be present at welds No. 3 and No. 4 as follows:

- Corrosive environment – Weld No. 3 would operate in a similar environment as weld No. 5 of the CRDM housing. Weld No. 4 would be exposed to a lower operating temperature than weld No. 5, however, TGSCC can still occur at 250 degrees Fahrenheit as evidenced by the Palisades previous operating experience with cracking identified in the seal housings that operate at even lower temperatures.

- Susceptible material – Welds No. 3 and No. 4 are composed of the same weld filler and base metal materials as weld No. 5 (e.g. weld filler material consistent with the type 316 stainless housing base metal). This material would be equally susceptible to TGSCC, as the type 347 stainless steel and weld filler materials used in the pre-2001 CRDM housing design that developed a through wall leak caused by TGSCC at weld No.3.

- Tensile stresses - While it is assumed that the corrective actions taken in response to the 2001 leak will reduce the potential for tensile stresses to exist on the inner surface of CRDM housings at welds No. 3 and No. 4, especially in light of the repairs made to welds No. 3 and No. 4, it had not been conclusively demonstrated that these tensile stresses have been eliminated. As such, it was not reasonable to conclude that tensile stresses were not present and, therefore, the potential for transgranular stress corrosion cracking had been eliminated.

Although the root cause report discussed manufacturing irregularities and misalignment between CRDM-24 and the support tube, seismic supports, and the associated reactor head penetration/CRDM nozzle as potential source of stresses leading to cracking, the root cause report also stated that "based on the lack of cracking found in the other 8 upper housings tested, the failed CRDM-24 upper housing contains an as-yet unidentified additional stress." Because the cause of the additional stress was not identified, the licensee had not established a sufficient basis in the RCR to exclude welds No. 3 and No. 4 from the extent of condition review (e.g. potential generic implications) This unknown additional stress as well as the propagation rate represent key differences as related to the cracking identified in 2001. The RCR documents the changes made to the CRDM housings in 2001 to reduce tensile stresses, but it does not document a justification for excluding welds No. 3 and No.4 from an evaluation for generic implications or corrective actions based on the results of the current root cause evaluation.

The inspectors identified that the licensee had not followed Procedure EN-LI-118 "Root Cause Evaluation," in the root cause review of the CRDM-24 leak as documented in report CR-PLP-2013-05623. Section 5.5 (12)e of EN-LI-118 required that the licensee "perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affects other SSC's." Additional details are provided in the procedure on how to conduct and document the evaluation. In this case, the inspectors identified that the licensee had not addressed or documented a basis in RCR CR-PLP-2013-05623 to exclude welds No. 3 and No. 4 from the generic factors discussed above that led to the 2012 leak in CRDM-24 (e.g. TGSCC at weld No. 5) sufficiently to meet the intent of the procedural requirements. The licensee entered this issue into the corrective action program as CR-PLP-2013-01500. To restore compliance with the procedure, the licensee intended to revise the inspection plan to add additional corrective actions to inspect a sample of welds No. 3 and No. 4 for TGSCC during the upcoming refueling outage.

Analysis: The inspectors determined that the failure to adequately evaluate and document the generic implications of the cause of the cracking identified in CRDM-24 in accordance with the root cause procedure EN-LI-118 was a performance deficiency that warranted a significance evaluation. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. The inspectors also answered "yes" to the More-than-Minor screening question "if left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, absent NRC identification, the licensee would not have completed further evaluations or inspections of CRDM housing welds which could have resulted in additional CRDM housing failure and leakage by TGSCC. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609, Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a large component rupture.

The inspectors determined that the primary cause of the failure to adequately consider welds No. 3 and No. 4 in the generic implications section of the root cause report related to the cross-cutting component of Human Performance, Decision Making, because licensee staff did not use conservative assumptions in decision making. Specifically, the licensee did not use conservative assumptions when excluding welds No. 3 and No. 4 as being susceptible to TGSCC and therefore include them in the generic implications section of the root cause report. (Item H.1(b) of IMC 310).

<u>Enforcement:</u> The inspectors identified a NCV of 10 CFR, Appendix B, Criterion V "Instructions, Procedures and Drawings", having a very low safety significance (Green), for failure to adequately evaluate and document the generic implications of the cause of cracking identified in CRDM-24 as it relates to welds No. 3 and No. 4 in accordance with the root cause procedure.

Title 10 CFR, Appendix B, Criterion V "Instructions, Procedures and Drawings requires in part, "Activities affecting quality shall be prescribed by documented procedures and shall be accomplished in accordance with these procedures."

Procedure EN-LI-118 "Root Cause Evaluation Process," Revision 17 states:

- 5.5 (12)e: perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affects other SSCs , organizations or work processes. Use the two step process in accordance with attachment 9.7
- Attachment 9.7: Determine whether the occurrence/consequence (problem) is isolated, or whether it has broader (generic or common mode) implications. Achieve this by asking the following questions:
    - i. Could this happen to equipment that is similar in function, design, or service condition?
    - ii. Could this happen to a group of components? (components of the same construction or materials that could be similarly affected by one condition)
- Attachment 9.7: Document the results of the above considerations. Include the following items in the write up:
    - i. Generic Implications (Is this problem/ cause limited to this component/equipment, or does it apply to others as well)
    - ii. Existing broader (generic/common mode) considerations

- 5.5(15)(10)c&f: Document proposed corrective actions and due dates to address valid generic implications. If no corrective action is recommended for a valid generic implication then document the basis for this conclusion and any risk or consequence identified as a result of taking no action.

Contrary to the above, from February 24, 2013 through April 18, 2013, the licensee failed to perform an activity affecting quality in accordance with procedure EN-LI-118. Specifically, the licensee did not evaluate and document the existing broader (generic/common mode) considerations associated with TGSCC at CRDM housing welds No. 3 and No. 4. Consequently, the licensee failed to propose corrective actions for the generic implications of TGSCC at CRDM housing welds No. 3 and No. 4 or to provide reasonable rationale why corrective actions were unnecessary. The licensee was considering adding welds No. 3 and No. 4 into its inspection plan for activities to be performed during the next refueling outage. Because of the very low safety significance and because the licensee entered this issue into their corrective action program (CR-PLP-2013-01500), it is being treated as a NCV consistent with Section 2.3.2 of the Enforcement Policy (NCV 05000255/2013003-xx).

4OA5 Other Activities

.1 (Closed) Unresolved Item 05000255/2012012-02: Potential Inadequate Degradation Evaluation of CRDM Housings

During a Special Inspection performed in August 2012, NRC inspectors identified an issue which could not be resolved without additional information (Unresolved Issue (URI)). This issue was associated with the rate of growth of the crack which created the through wall leak in CRDM-24, discovered on August 12, 2012. Identification of this crack growth rate is significant in determining appropriate intervals for future inspections to provide reasonable assurance that CRDM housing leakage will not recur.

Preliminary failure analysis data available at the time of the inspection indicated that the observed cracking was due to TGSCC. Cracking of this type is normally due to the presence of oxygen and chlorides at the location of the crack. When examining the fracture surface at the location the through-wall leak occurred, the licensee identified six concentric rings (beach marks) propagating in a radial direction from the ID out towards the OD of the housing. Beach marks are normally associated with fatigue failures and indicate the number of stress cycles from crack initiation to crack failure. In this case, there was no evidence that fatigue contributed to the failure. Despite the lack of evidence of fatigue, it was apparent that the crack which resulted in the CRDM-24 leak grew in increments. It was not, however, immediately apparent whether the increments were related to oxygen ingress (refueling outages) or temperature/pressure cycles (heatups/cooldowns).

At the time of the original inspection, 5 time intervals for through wall crack growth were under consideration. Two were based on literature crack growth data and three were based on interpretations of the beach marks. These time intervals were:

- Based on literature data, one contractor estimated that a 10% through wall flaw would require 4 years to reach 50% through wall.
- Based on literature data another contractor estimated the crack growth rate to be 2.1 x 10^{-5} in/hr or 0.18 in/yr. This is approximately three times faster than the crack growth rate proposed in the above mentioned rate.
- Based on the concept of oxygen ingress at refueling outages 6 cycles of 18 months duration would require 9 years for the crack to grow through wall
- Based on the concept of temperature/pressure cycles, the plant experienced 6 cold shutdowns in approximately 2 years preceding the crack. This equates to 2 years for the crack to grow through wall.
- Based on the concept that oxygen is required for crack growth and that oxygen is rapidly purged from the CRDM housings due to leakage past the seals, crack growth occurs only during the first few weeks of operation following a refueling outage, followed by no growth for the remaining period of operation when oxygen concentrations are low. This equates to 6 oxygen ingress events (irrespective of time between events) for the crack to grow through wall.

NRC inspectors including technical experts from NRC Headquarters performed a follow-up inspection to determine if the assumptions made by the licensee were conservative and the planned actions bounded those conservative assumptions. The inspectors reviewed a variety of documents associated with crack growth and inspection intervals. The inspectors noted the following statements included in the root cause report and vendor documents related to the determination of the appropriate crack growth rate:

- The laboratory conducting the failure analysis concluded, it could not be conclusively determined if the beach marks corresponded to refueling outages, (i.e., 18 month cycle) or shorter periods as occurred during outages over the past 24 months
- Palisades CRDM-21 leaked at weld No. 3 in 2001. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 failure. In calculating the crack growth rate of this crack, one contractor utilized an interval between beach marks which is much shorter than refueling outages. The intervals used are consistent with plant thermal cycles in which oxygen may or may not have been admitted into the CRDMs.
- A spare CRDM housing at Ft Calhoun leaked at weld No. 5 in 1990. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 Palisades failure. In calculating the crack growth rate of this crack, Ft Calhoun stated that the beach marks were related to refueling cycles. Ft Calhoun also performed calculations indicating that the oxygen level at the location of the flaw did not change with time (including in response to refueling outages) because the spare CRDM housing was not vented. Ft Calhoun's evaluation indicated that oxygen levels at the vicinity of the crack would have begun to decline through diffusion and convection had the intervals between outages been much longer than 18 months. This is interpreted to mean that the beach marks at Ft Calhoun are in response to pressure/thermal cycles.
- In at least one instance Palisades needed to repair the seals on a reactor coolant pump at a time other than an outage. This necessitates draining some of the water from the reactor coolant system and venting (admitting oxygen into) the CRDM housing. This represents an additional oxygen ingress event not included when determination of time to cracking is based on refueling outages.

- In its inspection plan, Palisades states that it will inspect all CRDM housings over the next 4 refueling outages, i.e., the interval between inspections is 1 refueling outage

Based on the above review, the inspection team noted that there were certain non conservative statements contained in the Root Cause Report and the inspection plan. These included:

- The crack growth rate based on refueling outages was understated. If oxygen ingress is related to beach marks, given the oxygen ingress event which occurred to repair reactor coolant pump seals, six beach marks would occur in a maximum of five refueling intervals rather than the six refueling intervals that were used to calculate the crack growth rate in the root cause report.
- The crack growth rate based on heat up and cool down cycles is overstated. The value in the root cause is based on 11 months. While 6 shutdowns did occur at the plant in 11 months several of these events did not result in pressure/temperature changes of the reactor coolant system. The appropriate time frame is 24 months rather than 11.
- The inspection plan contains a non conservative statement: "However, once the crack has been initiated it propagates over 4 to 5 operating cycles prior to going through wall." While this statement does reflect one of the proposed theories for crack growth, sufficient evidence to demonstrate reasonable assurance that this theory is correct, and thereby overcome the non-conservatism of this statement, was not provided.

Despite the existence of the non conservatisms stated above, the inspectors concluded:

- Sufficient evidence to conclusively determine the rate of crack growth does not exist.
- Crack growth based on pressure/temperature cycles is the most conservative of the potential crack growth mechanisms. In the absence of reasonable assurance of the correctness of less conservative mechanisms, through wall crack growth in two years must be utilized for regulatory purposes.
- The licensee has not formally committed to any of the crack growth mechanisms discussed.
- The licensee's inspection program includes inspection of all of the CRDM housings over the next 4 refueling outages. Approximately 25% of the housings will be inspected during each outage. The inspection of 25% of the CRDM housings each interval is sufficient to indicate that, in the event no indications are found during a given inspection, that the probability that flaws exist in other housings is extremely low. As such, it may be considered that the inspection of approximately 25% of the CRDM housings every refueling outage bounds all the crack growth rate mechanisms considered.

The inspectors considered this approach to inspection to be both acceptable and sufficient justification to close this URI.

4OA6 Management Meetings

.2 Interim Exit Meetings

An interim exit was conducted for:

- The results of the selected issue follow-up inspection, with Mr. T. Vitali, Site Vice President on April 18, 2013.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

B. Davis, Engineering Director
O. Gustafson, Licensing Manager
T. Foudy, Engineering Supervisor
B. Williams, Engineer
B. Dotson, Licensing

LIST OF ITEMS OPENED, CLOSED, DISCUSSED

Closed

05000255/2012012-01	URI	TS for PCS Pressure Boundary Leakage
05000255/2012012-02	URI	Potential Inadequate Degradation Evaluation of CRDM Housings
05000255/2012012-03	URI	Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality

Opened and Discussed

None.

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

4OA5 Other Activities

LIST OF ACRONYMS USED

RELEASE ENTIRETY

Giessner, John

From:	Giessner, John
Sent:	Monday, April 29, 2013 8:04 PM
To:	Sanchez Santiago, Elba; Hills, David; Alley, David
Cc:	Orth, Steven
Subject:	Palisades Input to DRP Report 2013 002 URI EMS _jbg.docx
Attachments:	Palisades Input to DRP Report 2013 002 URI EMS _jbg.docx

Overall in very good shape. See my comments to better word some stuff showing linkage with TS and crit V. Must show a good link

Also we need to hit a homerun on why the TS is a PD, but not willful _ as you know Lochbaum is really up on the verbiage so we need to say why it exists and why it is OK, but still a NCV – see my write-up.

Call me with questions.

C32

UNITED STATES
NUCLEAR REGULATORY COMMISSION
LISLE, IL 60532-4352

April XX, 2012

MEMORANDUM TO: Thomas Taylor
Senior Resident Inspector
Palisades Nuclear Plant

FROM: David Hills, Chief
Engineering Branch 3
Division of Reactor Safety

SUBJECT: PALISADES NUCLEAR PLANT
DRS INPUT TO INTEGRATED REPORT 05000255/2013002

Enclosed is the report input for the Palisades Nuclear Plant, Inspection Report 05000255/2013002. This report input documents completion of our review of Unresolved Items 05000255/2012012-01, "TS for PCS Pressure Boundary Leakage," 05000255/2012012-02, "Potential Inadequate Degradation Evaluation of CRDM Housings," and 05000255/2012012-03, "Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality." This report also completes one sample of the Problem Identification and Resolution, Selected Issue Follow-up inspection in accordance with IP 71152. I have reviewed this input to confirm compliance with Inspection Manual Chapter (IMC) 0612 and IMC 0305. This input is ready for inclusion into the integrated report and dissemination to the public.

Please input the following post Inspection Data into RPS:

Inspection Procedure	**Procedure Status** – see below: Incomplete, Complete, Complete by reference, Complete-full sample not available, Complete – opportunity to apply procedure not available, Not Applicable.	**Sample Size** – As documented in Scope Section If less than full sample size documented in the report input, the inspector must provide a justification below to enter into RPS and support the procedure status selected
71152	Complete	1

Inspection Report Item and Type (AV, FIN, NCV, URI or VIO)	**Cornerstone** (IE, MS, BI, EP, OR, PR, MISC)	**Cross Cutting Aspect** (H.n(i), P.n(i), S.n(i))	**Responsible Person/Owner**	**Procedure or TI** (71111.07T)	**RPS Branch Code (e.g. closeout responsibility)** EB1 3820, EB2 3870, EB3 3840, PST (RP) 3860, PSB (Safeguards) 3850, OB 3810
NCV-XXX	IE	n/a	E. Sanchez Santiago	71152	3820
NCV-XXX	IE	H.1(b)	E. Sanchez Santiago	71152	3820

Enclosure: Input to Inspection Report 05000255/2013002

cc w/encl: J. Giessner, Chief
C. Hernandez, Site Admin Assistant

CONTACT: E. Sanchez Santiago, DRS
(630) 829-9715

DOCUMENT NAME: G:\DRSIII\DRS\Work in Progress\-Palisades Input to DRP Report 2013 002 URI EMS.docx
9 Publicly Available 9 Non-Publicly Available 9 Sensitive 9 Non-Sensitive
To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

Field Code Changed

OFFICE	RIII		NRR		RIII		NRR	
NAME	ESanchezSantiago		DAlley		DHills		TLupold	
DATE	4/ /13							

OFFICIAL RECORD COPY

Cover Letter

X Green findings involving a violation were identified. Include the following:

Based on the results of this inspection, two NRC-identified findings of very low safety significance (Green) were identified. These findings were determined to involve a violation of NRC requirements. However, because of the very low safety significance and because the issues were entered into your corrective action program, the NRC is treating the issue as Non-Cited Violation, in accordance with Section 2.3.2 of the NRC Enforcement Policy.

TITLE PAGE

Inspectors: D. Alley, Senior Materials Engineer
E. Sanchez Santiago, Reactor Inspector

SUMMARY OF FINDINGS

A. NRC-Identified and Self-Revealed Findings

Cornerstones: Initiating Events

- Green. The inspectors identified a Finding with associated Non-Cited Violations (NCV) of 10 CFR Part 50, Appendix B, Criterion XVI and Technical Specification (TS) 3.4.13 Primary Coolant System (PCS) Operational Leakage Action B, for failure to prevent recurrence of CRDM cracking and leakage, a significant condition adverse to quality which resulted in a violation of TS. Specifically, for Criterion XVI the licensee failed to include the internal CRDM housing weld build-up area within the scope of corrective actions taken for ~~a~~ 2001 CRDM through wall leak on CRDM-21 ~~leakage event~~ caused by transgranular stress corrosion cracking (TGSCC). Subsequently ~~and consequently~~ a through wall leak recurred in the weld build-up area on ~~leakage recurred in~~ CRDM-24 in 2012 due to TGSCC. For TS 3.4.13, the licensee failed to shutdown in six hours for a pressure boundary leak as required by TS 3.14.13 Action B. The licensee replaced CRDM-24 upper housing and wrote CRXXXX.

  Comment [j1]: Self-revealed ? check old crite 16

  The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone objective to limit the likelihood of events that upset plant stability. The issue associated with the attribute of equipment performance. Specifically the licensee did not ~~limit the likelihood of events that upset plant stability by not taking~~ take adequate corrective actions to prevent recurrence of leakage in CRDM housings which represents a pressure boundary leakage. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System LOCA initiator contributor. The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June

19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the ~~inherent toughness (e.g. flaw tolerance)~~slow rate of change for leakage for this cracking mechanism and this type of material. ~~of the t~~Type 316 stainless steel material under TGSCC ~~such that~~will experience leakage rates well below a small break LOCA which would be observed through ~~inservice~~ the cracks, alerting operators ~~and~~ to take actions ~~taken~~ to ~~correct them~~shutdown the plant prior to experiencing a component rupture. ~~Despite the advanced age of the licensees decisions associated with this finding, the inspectors concluded that the finding was indicative of current performance.~~ The cause of this finding, non-conservative decision making occurred over ten years ago and is well outside the nominal 3 year period in IMC 0612; and would not be indicative of current performance, unless there were other opportunities to discover this specific issue. There were no recent opportunities; therefore, the inspectors concluded this was not indicative of current performance. ~~Specifically, the licensee more~~However more recently, the licensee, exhibited ~~similar~~ non-conservative decision making with respect to addressing the potential for CRDM housing cracking and leakage during the recent root cause (Section 4OA2.3 (b.2) of this report), ~~and~~ resulting in another finding. ~~However, given that both findings reflect upon the licensee's approach to basically the same equipment and technical issues, the inspectors did not apply a separate cross cutting aspect to this finding in that it is already~~This cross-cutting aspect will be captured through the other finding. (Section 4OA2.3(b.1))

Comment [j2]: Too technical

Comment [j3]: This PD itself was caused by old issues. We need to delete or reword as suggested.

- Green. The inspectors identified a Finding with an associated Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion V, for failure to accomplish quality activities in accordance with ~~follow~~ the prescribed procedures. ~~root cause procedure.~~ Specifically, the licensee failed to adequately evaluate and document the generic implications of the cause of the cracking identified in CRDM-24 in accordance with root cause procedure XXXXXX.

Comment [j4]: Corrective action

The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. The inspectors also answered "yes" to the More-than-Minor screening question "if left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, absent NRC identification, the licensee would not have completed further evaluations or inspections of CRDM housing welds which could have resulted in additional CRDM housing failure and leakage by TGSCC. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System LOCA initiator contributor. The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609, Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a

Comment [j5]: Same as above for attribute and cornerstone; and screening below for 316 steel

Comment [j6]: I'd just use if left uncorrected, but your call here

LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a large component rupture. The inspectors determined that the primary cause of the failure to adequately consider welds No. 3 and No. 4 in the generic implications section of the root cause report related to the cross-cutting component of Human Performance, Decision Making, because licensee staff did not use conservative assumptions in decision making. Specifically, the licensee did not use conservative assumptions when excluding welds No. 3 and No. 4 as being susceptible to TGSCC when there was not enough information to exclude them from consideration~~and therefore include them in the generic implications section of the root cause report~~. (Item H.1(b)). (Section 4OA2.3(b.2))

B. Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAILS

4. REACTOR SAFETY

4OA2 Identification and Resolution of Problems (71152)

.3 Selected Issue Follow-up Inspection: Through Wall Leakage of Control Rod Drive Mechanism (CRDM) Housing #24 (This inspection is part of the additional inspections included in the Palisades Deviation letter)

a. Inspection Scope

On August 12, 2012, the licensee shut down the plant to investigate an increase in unidentified leakage. The source of the leakage was determined to be a crack in CRDM-24. The NRC dispatched a special inspection team (SIT) to review the CRDM-24 leakage event. The results of that inspection are provided in Inspection Report 05000255/2012012. The licensee completed an evaluation to determine the cause of the cracking (CR-PLP-2012-05623).

From March 4, 2013 to March 15, 2013, the inspectors completed one inspection sample regarding problem identification and resolution based upon review of the licensee's root cause report contained in corrective action document CR-PLP-2012-05623. In addition the inspectors performed reviews related to three Unresolved Items (URI) identified during the SIT inspection:

- URI 05000255/2012012-01 TS for PCS Pressure Boundary Leakage. (The closure of this URI is documented in section 4OA2.3 (b.1) of this report.)
- URI 05000255/2012012-02 Potential Inadequate Degradation Evaluation of CRDM Housings (The closure of this URI is documented in section 4OA5.1 of this report)
- URI 05000255/2012012-03 Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality (The closure of this URI is documented in section 4OA2.3 (b.1) of this report.)

The inspectors reviewed the licensee's actions in accordance with performance attributes identified in IP 71152. Specifically, the inspectors reviewed licensee corrective action records to determine if: (1) the problems were accurately identified; (2) operability and reportability were adequately ascertained; (3) extent of condition and generic implications were appropriately addressed; (4) classification and prioritization of the problem were commensurate with safety significance; (5) root and contributing causes were identified; (6) corrective actions were appropriately focused to correct the problem; and (7) timely corrective actions were completed or proposed commensurate with the safety significance of the issues.

b. Findings

.1 Failure to Prevent Recurrence of CRDM Housing Cracking and Leakage

Introduction: The inspectors identified a Green Finding with associated Non-Cited Violations (NCV) of 10 CFR Part 50, Appendix B, Criterion XVI and Technical Specification (TS) 3.4.13 Primary Coolant System (PCS) Operational Leakage for failure to prevent recurrence of CRDM cracking and leakage, a significant condition adverse to quality which resulted in a violation of TS. Specifically, the licensee failed to include the internal CRDM housing weld build-up area within the scope of corrective actions taken for a 2001 CRDM leakage event and consequently leakage recurred in CRDM-24 in 2012.

Comment [j7]: Same as summary

Description: In 2001, the licensee discovered a steam leak in the housing of CRDM-21 caused by a through-wall TGSCC at CRDM housing weld No. 3 which was located just below the weld build-up region (weld No. 5). Weld No. 5 consists of a weld material deposit applied to the inside diameter (ID) of the CRDM housing which provides for alignment of the CRDM. This issue was categorized as a significant condition adverse to quality (SCAQ) by the licensee (CPAL0102186) and the licensee's root cause evaluation was documented in RCR/C-PAL-01-02186. The licensee considered this issue a SCAQ based on the procedure EN-LI-102 "Corrective Action Process" definition of "Conditions such as failures, malfunctions, deficiencies, deviations, defective material & equipment, and non-conformances which have resulted in, or could result in, a significant degradation or challenge to nuclear safety." The licensee concluded that the cracks in CRDM-21 were caused by TGSCC which occurred in areas of heavy grinding or machining tool marks. Specifically, this leak was the result of an ID initiated, axially oriented, transgranular crack in the austenitic stainless steel housing material. The failure analysis performed in response to this event identified both axial and circumferential cracks associated with weld No. 3. Extent of condition inspections revealed additional, non-through wall cracks associated with weld No. 3 in 41 of the 44 remaining housings for a total of 42 of 45 housings containing cracks.

Comment [j8]: Isn't this an Entergy procedure vs NMC; you'll need the NMC procedure or a statement saying it is similar

In response to the 2001cracking, Palisades replaced all 45 CRDM housings with housings thought to be more resistant to cracking. Principle changes included:

- Elimination of weld No. 2,
- Relocation of weld No. 3 to a higher location thereby minimizing the deposition of crud in the gap between the weld and the bottom plate of the rack and pinion assembly,
- Reduction in residual stresses and cold work on welds by requiring better surface finishes, and
- Use of heat sink welding to reduce ID residual tensile stresses.

Licensee corrective actions taken in response to the 2001 event were limited to butt welds. The inspectors reviewed the licensee actions to determine if they had been sufficient to eliminate one of the 3 necessary factors to cause TGSCC on the CRDM housings: (1) a susceptible material, (2) a corrosive environment and (3) tensile stress. The inspectors identified that the licensee had failed to eliminate one or more of the necessary factors at weld No. 5 (which was not a butt weld) to preclude TGSCC in the replacement housing. Specifically:

- The licensee's 2001 root cause report documented that weld No. 5 is exposed to essentially the same environment as the weld that experienced the cracking (corrosive environment remained unchanged).

- No analysis was completed on the stress conditions for weld No. ~~5 prior~~5 prior to approving the modified replacement housing design (the potential ~~for residual~~for residual tensile weld stresses on ID of CRDM surface was not ruled out by analysis and therefore, should have been considered).

- Fabrication restrictions to prohibit grinding were not applied to weld No. 5 (grinding promotes residual tensile stress state on ID of CRDM surface)

- Machining was performed on weld No. 5 during the fabrication process in order to achieve the dimensions and geometry specified in the design. This process induced cold work stresses in the weld.

- Material was changed from type 347 to type 316 stainless steel (both materials are essentially equally susceptible to TGSCC).

In January of 2002, an NRC special inspection team (SIT) (reference IR 50-2555/01-15) reviewed the licensee proposed corrective actions associated with the through-wall leakage of the CRDM-21 housing caused by TGSCC. The 2001 root cause report reviewed by the NRC stated the action to prevent recurrence was to "develop and implement an inspection plan to address areas and components identified in Attachment C-Extent of Condition". One of the components included in Attachment C was the CRD Mechanism. The recommended action was to perform volumetric inspection of the welds contained in the CRD Mechanism. Subsequently after the SIT a table of actions was compiled. The table also refers to a susceptibility analysis (EA-C-PAL-01-2186-02 "CRD Upper Housing and Nozzle Weld Susceptibility Comparison") to identify how degradation can be identified in this component. The objective of this document was to provide justification as to why the first weld (weld No. 1) above the reactor head is deemed to be less susceptible than the upper housing welds to failure by TGSCC and should not be included in the extent of condition. The susceptibility analysis excludes weld No. 5 because it is a weld overlay and not a butt weld and was deemed to be less susceptible to TGSCC than the butt welds. By not including weld No. 5 in the susceptibility analysis the licensee did not evaluate the stresses, material and environment of this weld to conclude it is not susceptible to TGSCC. An attachment to this analysis states machining marks were present on weld No. 5 which was identified as a key contributor to the cracking identified in weld No. 3. After this analysis was complete the licensee decided to replace all CRDM housings with the new design and control the fabrication process on the butt welds and the inspection plan would consist of the required ASME inspections. Weld No. 5 was excluded from these corrective actions and no fabrication controls were placed on weld No. 5 to reduce the stresses in this location.

Comment [j9]: What table? hard to follow, what existed when

On August 12, 2012, Palisades Nuclear ~~Power Station~~Plant was shut down to investigate an increase in unidentified leakage. During a walk-down performed post shutdown, the licensee discovered the source of the leakage to be a pressure boundary leak from CRDM-24. After further testing, the licensee determined the leak occurred because of a through-wall crack adjacent to weld No. 5.

The licensee formed a root cause team (RCT) staffed with licensee personnel and augmented with input from vendors. The root cause investigation was conducted in accordance with site procedure EN-LI-118 "Root Cause Evaluation Process" and was documented in root cause analysis report CR-PLP-2012-05623. In this report, the licensee's RCT determined that the probable cause of the cracking was:

> "Stresses in the weld build up area due to manufacturing irregularities and misalignments between CRDM-24 upper housing, support tube, and the associated reactor head penetration/CRDM nozzle. Based on lack of cracking found in the other 8 upper housings tested, the failed CRDM-24 upper housing contains an as-yet unidentified additional stress."

The RCT also identified the following contributing cause:

> "Transgranular Stress Corrosion Cracking (TGSCC) initiating within the internal weld build-up material of CRDM-24. The through wall crack initiated in the weld material and then propagated through the base metal until a leak developed in the outer diameter (OD) witness band region at the base of the ID weld build up.

This conclusion was based upon destructive and non destructive examinations (NDE) completed on a section of the failed housing which included the through-wall flaw. The RCT also relied upon vendor technical reports assessing the results of the NDE as well as vendor calculations related to the stresses in the CRDM housings.

To determine the extent of condition, the licensee performed ultrasonic (UT) examinations of weld No. 5 on eight additional CRDM housings. The licensee selected these housings based on being in a similar location on the head as CRDM-24, and previous cracking having been identified in some of these housings prior to the replacement of the CRDM upper housings and seal housings in 2002. The inspectors concluded that this was an adequate sample for an initial extent of condition review based upon the concept that, in light of eight negative exams, the statistical probability of a flaw in the remaining CRDM housings was very low. Additionally, the licensee planned to conduct examinations of more housings during the next refueling outage.

Based upon the recurrence of through-wall leakage in the CRDM housings caused by TGSCC, the inspectors concluded that the licensee actions were not adequate because the appropriate actions to preclude recurrence were within the licensee's ability to foresee and implement. Specifically, the inspectors concluded that the licensee did not effectively implement corrective actions for the 2001 CRDM housing leak resulting in the 2012 CRDM-24 housing leak. Also, in 1991, the Fort Calhoun plant had experienced through-wall leakage due to TGSCC at weld No. 5 of their CRDM housings (same housing design) and this operational experience had been reviewed by the licensee and dismissed. In the licensee's 2001 root cause evaluation, the licensee reviewed the weld build-up region failure by TGSCC at Fort Calhoun in the spare housing and concluded it would not occur at Palisades. This conclusion was based on the assumption that a higher oxygen environment (more aggressive environment) would exist in the spare Fort Calhoun housings than in the inservice Palisades housings. However the licensee did not confirm this assumption, nor did the licensee perform additional testing to determine if the environment of their inservice housings was sufficiently benign to prevent TGSCC. The licensee's 2012 RCT reached a similar conclusion and documented that due to organizational/ programmatic weakness at Palisades, the 1991 Fort Calhoun operating experience was not adequately utilized to include inspection of the weld No. 5. The inspectors identified that the licensee had missed a key opportunity to implement effective corrective actions that could have prevented recurrence of the 2001 leakage event and elected not to pursue. Specifically, in EA-EAR-2001-0426-01 the licensee considered fabricating the replacement housings with Inconel 600 material because it was much more resistant to TGSCC, but ultimately decided not to do so. Additionally, various vendor reports were generated related to this issue in the mid 2000's. Those reports documented the potential susceptibility of weld No. 5 to TGSCC due to their review of the CRDM housing conditions and available operating experience. The reports also noted that weld 5 was not inspected in any of the housing in 2001. One report in 2003 noted that weld #5 should have been examined as part of the action from the 2001 event since it was similar to Fort Calhoun. The issuance of these documents represents another opportunity for the licensee to identify the susceptibility of weld No. 5 to TGSCC prior to the cracking in CRDM-24.

The inspectors concluded the corrective actions taken in 2001 for an SCAQ on a CRDM for a through wall leak from TGSCC were not effective to preclude repetition. And another through wall leak did recur on a CRDM from TGSCC. This issues and it was within their ability to foresee and correct; therefore, the issue was a performance deficiency.

During the 2012 NRC special inspection, the NRC identified an unresolved item for the Technical Specification pressure boundary leak. LCO 3.4.13 subsection a does not allow any pressure boundary. Action B requires shutdown to mode 3 in 6 hours and mode 5 in 36 hours for such leakage. The licensee determined the CRDM-24 leakage commenced on or around July 14, 2012, and the plant continued to operate in this condition until August 12 2012. The NRC assessed this information and found it reasonable. The licensee failed to shutdown in six hours for a pressure boundary leak as required by TS 3.14.13 Action B. It should be noted the NRC previously assessed the site's action for rising unidentified leakage as part of the SIT. The NRC determined, at the time of higher unidentified leakage, the site took appropriate actions to attempt to locate the leak, eventually shutting down around .3 gallons per minute leakage (earlier than the TS value of 1 gpm value for unidentified leakage). The licensee did not know, specifically, of the through wall leakage until the shutdown on August 12, 2012 when a tour of previous inaccessible areas near the vessel head showed the leakage. Therefore, there were no willful aspects for this finding. Notwithstanding the site's previous actions, a TS violation did occur and the underlying reasons (part of the same finding Criteria XVI) were within their ability to foresee and correct; hence, a performance deficiency exists.

~~During the 2012 NRC special inspection, the NRC identified an unresolved item for the Technical Specification pressure boundary leak. The licensee determined the CRDM-24 leakage commenced on July 14, 2012 and the plant continued to operate in this condition, which was contrary to the TS 3.4.13 requirement of limiting PCS operational leakage to no pressure boundary leakage.~~

Based on the review discussed above, unresolved items 05000255/2012012-01 "TS for PCS Pressure Boundary Leakage" and 05000255/2012012-03 "Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality" are closed.

Analysis: The inspectors determined that the licensee's failure to prevent recurrence of TGSCC of the CRDM housings (a significant condition adverse to quality) that resulted in a violation of TS was a performance deficiency that warranted a significance evaluation. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. Specifically the licensee did not limit the likelihood of events that upset plant stability by not taking adequate corrective actions to prevent recurrence of leakage in CRDM housings which represents a pressure boundary leakage. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a component rupture.

Despite the advanced age of the licensees decisions associated with this finding, the inspectors concluded that the finding was indicative of current performance. Specifically, the licensee more recently exhibited similar non-conservative decision making with respect to addressing the potential for CRDM housing cracking and leakage (Section 4OA2.3 (b.2) of this report) and resulting in another finding. However, given that both findings reflect upon the licensee's approach to basically the same equipment and technical issues, the inspectors did not apply a separate cross cutting aspect to this finding in that it is already captured through the other finding.

Comment [j10]: Comments from summary paragraph

Enforcement: During the inspection, the inspectors identified two non-cited violations of NRC requirements. ~~The inspectors identified NCVs of 10 CFR, Appendix B, Criterion XVI "Corrective Actions", and Technical Specification 3.4.13 "Primary Coolant System Operational Leakage", having a very low safety significance (Green), for failure to prevent the recurrence of leakage in CRDM housings due to TGSCC resulting in the operation of the reactor with pressure boundary leakage, a condition prohibited by TS. Given that both violations relate to the same performance deficiency, they are considered as one finding.~~

Title 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that, for significant conditions adverse to quality, the cause of the condition is determined and corrective action taken to preclude repetition.

Technical Specifications 3.4.13 requires PCS operational leakage be limited to "No pressure boundary LEAKAGE" when in Modes 1 through 4. Required Action B of the TS requires the licensee to be mode 3 in 6 hours Completion Time, and be mode 5 in 36 hours Completion Time for pressure boundary leakage.

Contrary to the above, as of August 12, 2012, the licensee had failed to take corrective actions to preclude repetition for a significant condition adverse to quality. Specifically, June 21, 2001, the licensee discovered a through wall leak in CRDM-21 due to TGSCC and failed to reasonably include weld No. 5 in the corrective actions which resulted in a subsequent through wall leak in CRDM-24 due to TGSCC.

Contrary to the above on or around July 14, 2012, ~~The~~ pressure boundary leakage at CRDM-24 ~~began~~existed ~~on July 14, 2012~~, and the licensee failed to take the TS Required Action to be mode 3 and be in mode 5, in the TS Completion Times of 6 hours 36 hours respectively. ~~the~~ The plant continued to operate until August 12, 2012~~,~~. ~~which is contrary to the TS requirement of limiting operational leakage to no pressure boundary leakage.~~

~~As a result of the second through wall leak, the licensee took corrective actions which included the development of an inspection plan that would inspect weld No. 5 every outage until all CRDM housings were inspected.~~

Because these violations were of very low safety significance, were not willful, and were entered into the licensee's corrective action program as CR-PLP-2013-01134, these violations are being treated as an NCVs, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000255/2013002-*xx*; Failure to Prevent Recurrence of a Significant Condition Adverse to Quality).

.2 Failure to Adequately Address the Generic Implications of the Cracking identified in CRDM 24

Introduction: The inspectors identified a Green Finding with an associated Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion V, for failure to follow the root cause procedure. Specifically, the licensee failed to adequately evaluate and document the generic implications of the cause of the cracking identified in CRDM-24.

Comment [j11]: See previous stuff

Description: As a result of the cracking identified in CRDM-24, which was characterized as a SCAQ, the licensee performed a root cause evaluation in accordance with procedure EN-LI-118, "Root Cause Evaluation". This procedure is identified as quality related and serves to implement a portion of the licensee's quality assurance program. While reviewing the 2012 root cause report CR-PLP-2013-05623 related to the cracking identified in CRDM-24, generated as a result of the root cause evaluation, the inspectors identified that the licensee had not appropriately considered the generic implications of the cracking in the extent of condition review. The licensee's proposed corrective actions narrowly focused on weld No. 5, instead of also including broader actions to ensure other CRDM housing welds were fit for their intended service life.

On March 13, 2013, the inspectors requested that the licensee provide the bases for excluding other CRDM housing welds (weld No. 3 below weld No. 5 and weld No. 4 above weld No. 5) from the scope of planned corrective actions. On March 29, the licensee provided additional information to justify excluding these welds from the scope of the corrective actions. The licensee credited the corrective actions associated with the modifications to the CRDM housing design completed in 2001 as the basis to exclude housing welds No. 3 and No. 4 from additional actions to identify the extent of TGSCC. The corrective actions taken in 2001 included performing heat sink welding, which is a methodology used to reduce the stresses on the inner ID of the weld. The licensee also changed the design to reduce design stresses at weld No. 3 and specified a smoother surface finish (RMS 125) to reduce potential crack initiation points. The licensee stated that these actions would produce compressive stresses on the ID of welds No. 3 and No. 4 making them immune from cracking. The inspectors acknowledged that these actions would reduce the tensile stress at the ID surface and thus reduce the probability of initiating TGSCC.

However, the information provided did not demonstrate that TGSCC would not occur because, it did not demonstrate that tensile stress would be eliminated at the ID surface during operation. In particular, repairs completed at the inner surface of weld No. 4, would result in high residual tensile stress at the inside surface of the weld which would promote the initiation of TGSCC. Repairs were also performed on weld No. 3 from the OD surface of the weld. The licensee believed that the last pass heat sink welding process would be sufficient to ensure residual compressive stress would remain at the ID surface of Weld No. 3 even with repairs to the OD surface. However, the licensee had not completed detailed residual weld stress testing or modeling to confirm this assumption.

The inspectors identified that the three factors required for TGSCC could still be present at welds No. 3 and No. 4 as follows:

- Corrosive environment – Weld No. 3 would operate in a similar environment as weld No. 5 of the CRDM housing. Weld No. 4 would be exposed to a lower operating temperature than weld No. 5, however, TGSCC can still occur at 250 degrees Fahrenheit as evidenced by the Palisades previous operating experience with cracking identified in the seal housings that operate at even lower temperatures.

- Susceptible material – Welds No. 3 and No. 4 are composed of the same weld filler and base metal materials as weld No. 5 (e.g. weld filler material consistent with the type 316 stainless housing base metal). This material would be equally susceptible to TGSCC, as the type 347 stainless steel and weld filler materials used in the pre-2001 CRDM housing design that developed a through wall leak caused by TGSCC at weld No.3.

- Tensile stresses - While it is assumed that the corrective actions taken in response to the 2001 leak will reduce the potential for tensile stresses to exist on the inner surface of CRDM housings at welds No. 3 and No. 4, especially in light of the repairs made to welds No. 3 and No. 4, it had not been conclusively demonstrated that these tensile stresses have been eliminated. As such, it was not reasonable to conclude that tensile stresses were not present and, therefore, the potential for transgranular stress corrosion cracking had been eliminated.

Although the root cause report discussed manufacturing irregularities and misalignment between CRDM-24 and the support tube, seismic supports, and the associated reactor head penetration/CRDM nozzle as potential source of stresses leading to cracking, the root cause report also stated that "based on the lack of cracking found in the other 8 upper housings tested, the failed CRDM-24 upper housing contains an as-yet unidentified additional stress." Because the cause of the additional stress was not identified, the licensee had not established a sufficient basis in the RCR to exclude welds No. 3 and No. 4 from the extent of condition review (e.g. potential generic implications). This unknown additional stress as well as the propagation rate represent key differences as related to the cracking identified in 2001. The RCR documents the changes made to the CRDM housings in 2001 to reduce tensile stresses, but it does not document a justification for excluding welds No. 3 and No.4 from an evaluation for generic implications or corrective actions based on the results of the current root cause evaluation.

The inspectors identified that the licensee had not followed Procedure EN-LI-118 "Root Cause Evaluation," in the root cause review of the CRDM-24 leak as documented in report CR-PLP-2013-05623. Section 5.5 (12)e of EN-LI-118 required that the licensee "perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affects other SSC's." Additional details are provided in the procedure on how to conduct and document the evaluation. In this case, the inspectors identified that the licensee had not addressed or documented a basis in RCR CR-PLP-2013-05623 to exclude welds No. 3 and No. 4 from the generic factors discussed above that led to the 2012 leak in CRDM-24 (e.g. TGSCC at weld No. 5) ~~sufficiently~~ to meet ~~the intent of~~ the procedural requirements. The licensee entered this issue into the corrective action program as CR-PLP-2013-01500. To restore compliance with the procedure, the licensee intended to revise the inspection plan to add additional corrective actions to inspect a sample of welds No. 3 and No. 4 for TGSCC during the upcoming refueling outage.

Analysis: The inspectors determined that the failure to adequately evaluate and document the generic implications of the cause of the cracking identified in CRDM-24 in accordance with the root cause procedure EN-LI-118 was a performance deficiency that warranted a significance evaluation. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. The inspectors also answered "yes" to the More-than-Minor screening question "if left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, absent NRC identification, the licensee would not have completed further evaluations or inspections of CRDM housing welds which could have resulted in additional CRDM housing failure and leakage by TGSCC. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

Comment [j12]: Cornerstone/ attribute info as in summry

The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609, Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function.

The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a large component rupture.

The inspectors determined that the primary cause of the failure to adequately consider welds No. 3 and No. 4 in the generic implications section of the root cause report related to the cross-cutting component of Human Performance, Decision Making, because licensee staff did not use conservative assumptions in decision making. Specifically, the licensee did not use conservative assumptions when excluding welds No. 3 and No. 4 as being susceptible to TGSCC and therefore include them in the generic implications section of the root cause report. (Item H.1(b) of IMC 310).

Enforcement: During the inspection, the inspectors identified one non-cited violations of NRC requirements:~~The inspectors identified a NCV of 10 CFR, Appendix B, Criterion V "Instructions, Procedures and Drawings", having a very low safety significance (Green), for failure to adequately evaluate and document the generic implications of the cause of cracking identified in CRDM 24 as it relates to welds No. 3 and No. 4 in accordance with the root cause procedure.~~

Title 10 CFR, Appendix B, Criterion V "Instructions, Procedures and Drawings requires in part, "Activities affecting quality shall be prescribed by documented procedures and shall be accomplished in accordance with these procedures."

Comment [j13]: Only quaote verbatim stuff

Procedure EN-LI-118 "Root Cause Evaluation Process," Revision 17 states:

- 5.5 (12)e: perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affects other SSCs , organizations or work processes. Use the two step process in accordance with attachment 9.7
- Attachment 9.7: Determine whether the occurrence/consequence (problem) is isolated, or whether it has broader (generic or common mode) implications. Achieve this by asking the following questions:
    i. Could this happen to equipment that is similar in function, design, or service condition?
    ii. Could this happen to a group of components? (components of the same construction or materials that could be similarly affected by one condition)
- Attachment 9.7: Document the results of the above considerations. Include the following items in the write up:
    i. Generic Implications (Is this problem/ cause limited to this component/equipment, or does it apply to others as well)
    ii. Existing broader (generic/common mode) considerations
- 5.5(15)(10)c&f: Document proposed corrective actions and due dates to address valid generic implications. If no corrective action is recommended for a valid generic implication then document the basis for this conclusion and any risk or consequence identified as a result of taking no action.

Contrary to the above, from February 24, 2013 through April 18, 2013, the licensee failed to ~~perform~~ accomplish ~~an activity~~ activities as prescribed in ~~a~~ ~~affecting~~ quality ~~in~~

~~accordance with~~ procedure EN-LI-118. Specifically, the licensee failed to accomplish steps XXXXX by did not ~~evaluate~~ fully evaluating and documenting the existing broader (generic/common mode) considerations, extent of condition/cause associated with TGSCC at CRDM housing welds No. 3 and No. 4. Consequently, the licensee failed to propose corrective actions for the generic implications of TGSCC at CRDM housing welds No. 3 and No. 4 or to provide reasonable rationale why corrective actions were unnecessary. ~~The licensee was considering adding welds No. 3 and No. 4 into its inspection plan for activities to be performed during the next refueling outage.~~ Because of the very low safety significance and because the licensee entered this issue into their corrective action program (CR-PLP-2013-01500), it is being treated as a NCV consistent with Section 2.3.2 of the Enforcement Policy (NCV 05000255/2013003-xx).

Comment [j14]: Help me with the steps

4OA5 Other Activities

.1 (Closed) Unresolved Item 05000255/2012012-02: Potential Inadequate Degradation Evaluation of CRDM Housings (This inspection is part of the additional inspections included in the Palisades Deviation letter)

During a Special Inspection performed in August 2012, NRC inspectors identified an issue which could not be resolved without additional information (Unresolved Issue (URI)). This issue was associated with the rate of growth of the crack which created the through wall leak in CRDM-24, discovered on August 12, 2012. Identification of this crack growth rate is significant in determining appropriate intervals for future inspections to provide reasonable assurance that CRDM housing leakage will not recur.

Preliminary failure analysis data available at the time of the inspection indicated that the observed cracking was due to TGSCC. Cracking of this type is normally due to the presence of oxygen and chlorides at the location of the crack. When examining the fracture surface at the location the through-wall leak occurred, the licensee identified six concentric rings (beach marks) propagating in a radial direction from the ID out towards the OD of the housing. Beach marks are normally associated with fatigue failures and indicate the number of stress cycles from crack initiation to crack failure. In this case, there was no evidence that fatigue contributed to the failure. Despite the lack of evidence of fatigue, it was apparent that the crack which resulted in the CRDM-24 leak grew in increments. It was not, however, immediately apparent whether the increments were related to oxygen ingress (refueling outages) or temperature/pressure cycles (heatups/cooldowns).

At the time of the original inspection, 5 time intervals for through wall crack growth were under consideration. Two were based on literature crack growth data and three were based on interpretations of the beach marks. These time intervals were:

- Based on literature data, one contractor estimated that a 10% through wall flaw would require 4 years to reach 50% through wall.
- Based on literature data another contractor estimated the crack growth rate to be 2.1 x 10^{-5} in/hr or 0.18 in/yr. This is approximately three times faster than the crack growth rate proposed in the above mentioned rate.
- Based on the concept of oxygen ingress at refueling outages 6 cycles of 18 months duration would require 9 years for the crack to grow through wall.

- Based on the concept of temperature/pressure cycles, the plant experienced 6 cold shutdowns in approximately 2 years preceding the crack. This equates to 2 years for the crack to grow through wall.
- Based on the concept that oxygen is required for crack growth and that oxygen is rapidly purged from the CRDM housings due to leakage past the seals, crack growth occurs only during the first few weeks of operation following a refueling outage, followed by no growth for the remaining period of operation when oxygen concentrations are low. This equates to 6 oxygen ingress events (irrespective of time between events) for the crack to grow through wall.

NRC inspectors, including technical experts from NRC Headquarters, performed a follow-up inspection to determine if the assumptions made by the licensee were conservative and the planned actions bounded those conservative assumptions. The inspectors reviewed a variety of documents associated with crack growth and inspection intervals. The inspectors noted the following statements included in the root cause report and vendor documents related to the determination of the appropriate crack growth rate:

- The laboratory conducting the failure analysis concluded, it could not be conclusively determined if the beach marks corresponded to refueling outages, (i.e., 18 month cycle) or shorter periods as occurred during outages over the past 24 months
- Palisades CRDM-21 leaked at weld No. 3 in 2001. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 failure. In calculating the crack growth rate of this crack, one contractor utilized an interval between beach marks which is much shorter than refueling outages. The intervals used are consistent with plant thermal cycles in which oxygen may or may not have been admitted into the CRDMs.
- A spare CRDM housing at Ft Calhoun leaked at weld No. 5 in 1990. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 Palisades failure. In calculating the crack growth rate of this crack, Ft Calhoun stated that the beach marks were related to refueling cycles. Ft Calhoun also performed calculations indicating that the oxygen level at the location of the flaw did not change with time (including in response to refueling outages) because the spare CRDM housing was not vented. Ft Calhoun's evaluation indicated that oxygen levels at the vicinity of the crack would have begun to decline through diffusion and convection had the intervals between outages been much longer than 18 months. This is interpreted to mean that the beach marks at Ft Calhoun are in response to pressure/thermal cycles.
- In at least one instance Palisades needed to repair the seals on a reactor coolant pump at a time other than an outage. This necessitates draining some of the water from the reactor coolant system and venting (admitting oxygen into) the CRDM housing. This represents an additional oxygen ingress event not included when determination of time to cracking is based on refueling outages.
- In its inspection plan, Palisades states that it will inspect all CRDM housings over the next 4 refueling outages, i.e., the interval between inspections is 1 refueling outage

Based on the above review, the inspection team noted that there were certain non conservative statements contained in the Root Cause Report and the inspection plan. These included:

- The crack growth rate based on refueling outages was understated. If oxygen ingress is related to beach marks, given the oxygen ingress event which occurred to repair reactor coolant pump seals, six beach marks would occur in a maximum of five refueling intervals rather than the six refueling intervals that were used to calculate the crack growth rate in the root cause report.
- The crack growth rate based on heat up and cool down cycles is overstated. The value in the root cause is based on 11 months. While 6 shutdowns did occur at the plant in 11 months, several of these events did not result in pressure/temperature changes of the reactor coolant system. The appropriate time frame is 24 months rather than 11.
- The inspection plan contains a non conservative statement: "However, once the crack has been initiated it propagates over 4 to 5 operating cycles prior to going through wall." While this statement does reflect one of the proposed theories for crack growth, sufficient evidence to demonstrate reasonable assurance that this theory is correct, and thereby overcome the non-conservatism of this statement, was not provided.

Despite the existence of the non conservatisms stated above, the inspectors concluded:

- Sufficient evidence to conclusively determine the rate of crack growth does not exist.
- Crack growth based on pressure/temperature cycles is the most conservative of the potential crack growth mechanisms. In the absence of reasonable assurance of the correctness of less conservative mechanisms, through wall crack growth in two years must be utilized for regulatory purposes.
- The licensee has not formally committed to any of the crack growth mechanisms discussed.
- The licensee's inspection program includes inspection of all of the CRDM housings over the next 4 refueling outages. Approximately 25% of the housings will be inspected during each outage. The inspection of 25% of the CRDM housings each interval is sufficient to indicate that, in the event no indications are found during a given inspection, that the probability that flaws exist in other housings is extremely low. As such, it may be considered that the inspection of approximately 25% of the CRDM housings every refueling outage bounds all the crack growth rate mechanisms considered.

Overall, some weaknesses did exist in the site's assessment, but none of these issues arose above the level of a minor performance deficiency for the evaluations completed. With the corrective actions in place to monitor the CRDMs, ~~T~~the inspectors considered this approach to inspection to be both acceptable and sufficient justification to close this URI.

4OA6 Management Meetings

.2 Interim Exit Meetings

An interim exit was conducted for:

- The results of the selected issue follow-up inspection, with Mr. T. Vitali, Site Vice President on April 18, 2013.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

B. Davis, Engineering Director
O. Gustafson, Licensing Manager
T. Foudy, Engineering Supervisor
B. Williams, Engineer
B. Dotson, Licensing

LIST OF ITEMS OPENED, CLOSED, DISCUSSED

Closed

05000255/2012012-01	URI	TS for PCS Pressure Boundary Leakage
05000255/2012012-02	URI	Potential Inadequate Degradation Evaluation of CRDM Housings
05000255/2012012-03	URI	Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality

Opened and Discussed

None.

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

4OA5 Other Activities

LIST OF ACRONYMS USED

Sanchez Santiago, Elba

From: Hills, David
Sent: Tuesday, April 30, 2013 12:59 PM
To: Giessner, John
Cc: Sanchez Santiago, Elba
Subject: RE: Palisades input (CRDM inspection)

Categories: FOIA

Maybe you are right with respect to calling the first finding not current performance. By saying it is indicative of current performance and having the two findings share a single cross cutting, we kind of help make Steve's argument for him. Better to keep the two findings as separated as possible. In addition to addressing Steve's comment by better differentiating the two findings from one another as indicated below, maybe we should just say no cross cutting for the first findng because of the reasons you state.

And I agree that shoehorning them would complicate the independent review. On the other hand, somewhat less likely they will dispute if all one finding. What would be the point? Even if they win on one violation/issue, they still have one finding with one PD and one cross cutting, same as before.

But I can go other way, same as I could have gone with finding and no violation on the last one. All viable options in my mind. Elba's opinion weighs heavily in whatever direction we go I think.

- Dave

From: Giessner, John
Sent: Tuesday, April 30, 2013 12:42 PM
To: Hills, David; Sanchez Santiago, Elba
Subject: RE: Palisades input (CRDM inspection)

Elba's ease is key
I see the 2001 distinct and older (same causal factor but different people/ process- not current performcne) – it occurred and they did stuff to fix – there were some opportunities on the way to see weld #5 (vendor reports) they missed it – it is in their ability to foresee and correct.

In 2012 when it now recurs they should have gone back in the EOC/EOCa and look at the root and extents and say what did we learn. They missed key items. Perhaps you are right – have the 2nd finding have less focus on the past – other than to say they did x and y, but in their relook – did not do z.

Also if we shoehorned them all and they dispute it – it's a big can of worms --

---

From: Hills, David
Sent: Tuesday, April 30, 2013 12:32 PM
To: Giessner, John; Sanchez Santiago, Elba
Subject: RE: Palisades input (CRDM inspection)

Depends on how you define the PD. Right now the way we have it, the first finding's PD isn't a single act in time. It is the lack of actions over time including missed opportunities. We would just need to extent that same PD a little longer in time to also cover recent actions and revise the wording a little to be more encompassing.

C314

Other option to satisfy Steve, of course, if Elba thinks she can do it, is to reword to make the second finding new causes and corrective actions stand out more and to specifically differentiate them from those of the first finding (how are they different/direct comparison). In fact, the new causes are the same as the old causes (hence why they share the same cross cutting), just the associated actions (or lack thereof) are arguably different and separate in time. Would need to address that somehow.

Which would be easier for Elba?

From: Giessner, John
Sent: Tuesday, April 30, 2013 12:16 PM
To: Hills, David; Sanchez Santiago, Elba
Subject: RE: Palisades input (CRDM inspection)

They are a different PD's. One is for what happened in 2001, one for 2012. Too hard to shoehorn.

I think

From: Hills, David
Sent: Tuesday, April 30, 2013 12:13 PM
To: Sanchez Santiago, Elba; Giessner, John
Subject: FW: Palisades input (CRDM inspection)

One other option to alleviate Steve's concern noted below. Presently, we have one finding with two violations and a second finding with one violation. We could combine them into one finding with three violations. Performance deficiency would be inadequate actions over an extended period (including recent) to address CRDM cracking and leakage. That would address Steve's double jeopardy concern I think. Plus would make the one cross cutting for two findings become one cross cutting for one finding which is a little less ackward and more straightforward I think. Thoughts?

- Dave

From: Orth, Steven
Sent: Tuesday, April 30, 2013 11:40 AM
To: Sanchez Santiago, Elba
Cc: Hills, David; Giessner, John
Subject: RE: Palisades input (CRDM inspection)

Elba,

Attached are my comments and questions.

In particular, our basis for the Criterion V violation is not clearly documented. I still don't understand what NEW causes and corrective actions should be considered for the remaining welds (3 &4). My read is that we need to separate from previous RCE and corrective actions from 2001. Otherwise, we're just hitting them again for problems in 2001.

Steve

From: Sanchez Santiago, Elba
Sent: Monday, April 29, 2013 12:31 PM
To: Hills, David; Orth, Steven; Giessner, John
Subject: Palisades input (CRDM inspection)

All,

Attached is the latest revision to the input to the Palisades quarterly report. I included the changes recommended as a result of our discussions. Please review and let me know if you have any comments or questions.

Thanks,

Elba M. Sanchez Santiago
Reactor Engineer
RIII/ DRS/ EB1
630-829-9715

UNITED STATES
NUCLEAR REGULATORY COMMISSION
LISLE, IL 60532-4352

April XX, 2012

MEMORANDUM TO: Thomas Taylor
Senior Resident Inspector
Palisades Nuclear Plant

FROM: David Hills, Chief
Engineering Branch 3
Division of Reactor Safety

SUBJECT: PALISADES NUCLEAR PLANT
DRS INPUT TO INTEGRATED REPORT 05000255/2013002

Enclosed is the report input for the Palisades Nuclear Plant, Inspection Report 05000255/2013002. This report input documents completion of our review of Unresolved Items 05000255/2012012-01, "TS for PCS Pressure Boundary Leakage," 05000255/2012012-02, "Potential Inadequate Degradation Evaluation of CRDM Housings," and 05000255/2012012-03, "Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality." This report also completes one sample of the Problem Identification and Resolution, Selected Issue Follow-up inspection in accordance with IP 71152. I have reviewed this input to confirm compliance with Inspection Manual Chapter (IMC) 0612 and IMC 0305. This input is ready for inclusion into the integrated report and dissemination to the public.

Please input the following post Inspection Data into RPS:

Inspection Procedure	**Procedure Status** – see below: Incomplete, Complete, Complete by reference, Complete-full sample not available, Complete – opportunity to apply procedure not available, Not Applicable.	**Sample Size** – As documented in Scope Section If less than full sample size documented in the report input, the inspector must provide a justification below to enter into RPS and support the procedure status selected
71152	Complete	1

Inspection Report Item and Type (AV, FIN, NCV, URI or VIO)	**Cornerstone** (IE, MS, BI, EP, OR, PR, MISC)	**Cross Cutting Aspect** (H.n(i), P.n(i), S.n(i))	**Responsible Person/Owner**	**Procedure or TI** (71111.07T)	**RPS Branch Code (e.g. closeout responsibility)** EB1 3820 EB2 3870 EB3 3840 PST (RP) 3860 PSB (Safeguards) 3850 OB 3810
NCV-XXX	IE	n/a	E. Sanchez Santiago	71152	3820
NCV-XXX	IE	H.1(b)	E. Sanchez Santiago	71152	3820

Enclosure: Input to Inspection Report 05000255/2013002

cc w/encl: J. Giessner, Chief
C. Hernandez, Site Admin Assistant

CONTACT: E. Sanchez Santiago, DRS
(630) 829-9715

DOCUMENT NAME: G:\DRSIII\DRS\Work in Progress\-Palisades Input to DRP Report 2013 002 URI EMS.docx
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Field Code Changed

OFFICE	RIII		NRR		RIII		NRR	
NAME	ESanchezSantiago		DAlley		DHills		TLupold	
DATE	4/ /13							

OFFICIAL RECORD COPY

Cover Letter

X Green findings involving a violation were identified. Include the following:

Based on the results of this inspection, two NRC-identified findings of very low safety significance (Green) were identified. These findings were determined to involve a violation of NRC requirements. However, because of the very low safety significance and because the issues were entered into your corrective action program, the NRC is treating the issue as Non-Cited Violation, in accordance with Section 2.3.2 of the NRC Enforcement Policy.

TITLE PAGE

Inspectors: D. Alley, Senior Materials Engineer
E. Sanchez Santiago, Reactor Inspector

SUMMARY OF FINDINGS

A. NRC-Identified and Self-Revealed Findings

Cornerstones: Initiating Events

- Green. The inspectors identified a Finding with associated Non-Cited Violations (NCV) of 10 CFR Part 50, Appendix B, Criterion XVI and Technical Specification (TS) 3.4.13 Primary Coolant System (PCS) Operational Leakage for the failure to take corrective actions to prevent recurrence of CRDM cracking and leakage, a significant condition adverse to quality which resulted in a violation of TS. Specifically, the licensee failed to include the internal CRDM housing weld build-up area within the scope of corrective actions taken for a 2001 CRDM leakage event and consequently leakage recurred in CRDM-24 in 2012.

  The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. Specifically the licensee did not limit the likelihood of events that upset plant stability by not taking adequate corrective actions to prevent recurrence of leakage in CRDM housings which represents a pressure boundary leakage. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System LOCA initiator contributor. The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g.

flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a component rupture. Despite the advanced age of the licensees decisions associated with this finding, the inspectors concluded that the finding was indicative of current performance. Specifically, the licensee more recently exhibited similar non-conservative decision making with respect to addressing the potential for CRDM housing cracking and leakage (Section 4OA2.3 (b.2) of this report) and resulting in another finding. However, given that both findings reflect upon the licensee's approach to basically the same equipment and technical issues, the inspectors did not apply a separate cross cutting aspect to this finding in that it is already captured through the other finding. (Section 4OA2.3(b.1))

- Green. The inspectors identified a Finding with an associated Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion V, for failure to follow the root cause procedure. Specifically, the licensee failed to adequately evaluate and document the generic implications of the cause of the cracking identified in CRDM-24.

  The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. The inspectors also answered "yes" to the More-than-Minor screening question "if left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, absent NRC identification, the licensee would not have completed further evaluations or inspections of CRDM housing welds which could have resulted in additional CRDM housing failure and leakage by TGSCC. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System LOCA initiator contributor. The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609, Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a large component rupture. The inspectors determined that the primary cause of the failure to adequately consider welds No. 3 and No. 4 in the generic implications section of the root cause report related to the cross-cutting component of Human Performance, Decision Making, because licensee staff did not use conservative assumptions in decision making. Specifically, the licensee did not use conservative assumptions when excluding welds No. 3 and No. 4 as being susceptible to TGSCC and therefore include them in the generic implications section of the root cause report. (Item H.1(b)). (Section 4OA2.3(b.2))

B. **Licensee-Identified Violations**

No violations of significance were identified.

REPORT DETAILS

4. REACTOR SAFETY

4OA2 Identification and Resolution of Problems (71152)

.3 Selected Issue Follow-up Inspection: Through Wall Leakage of Control Rod Drive Mechanism (CRDM) Housing #24

a. Inspection Scope

On August 12, 2012, the licensee shut down the plant to investigate an increase in unidentified leakage. The source of the leakage was determined to be a crack in CRDM-24. The NRC dispatched a special inspection team (SIT) to review the CRDM-24 leakage event. The results of that inspection are provided in Inspection Report 05000255/2012012. The licensee completed an evaluation to determine the cause of the cracking (CR-PLP-2012-05623).

From March 4, 2013 to March 15, 2013, the inspectors completed one inspection sample regarding problem identification and resolution based upon review of the licensee's root cause report contained in corrective action document CR-PLP-2012-05623. In addition the inspectors performed reviews related to three Unresolved Items (URI) identified during the SIT inspection:

- URI 05000255/2012012-01 TS for PCS Pressure Boundary Leakage. (The closure of this URI is documented in section 4OA2.3 (b.1) of this report.)
- URI 05000255/2012012-02 Potential Inadequate Degradation Evaluation of CRDM Housings (The closure of this URI is documented in section 4OA5.1 of this report)
- URI 05000255/2012012-03 Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality (The closure of this URI is documented in section 4OA2.3 (b.1) of this report.)

The inspectors reviewed the licensee's actions in accordance with performance attributes identified in IP 71152. Specifically, the inspectors reviewed licensee corrective action records to determine if: (1) the problems were accurately identified; (2) operability and reportability were adequately ascertained; (3) extent of condition and generic implications were appropriately addressed; (4) classification and prioritization of the problem were commensurate with safety significance; (5) root and contributing causes were identified; (6) corrective actions were appropriately focused to correct the problem; and (7) timely corrective actions were completed or proposed commensurate with the safety significance of the issues.

b. Findings

.1 Failure to Take Corrective Actions to Prevent Recurrence of CRDM Housing Cracking and Leakage

Introduction: The inspectors identified a Green Finding with associated Non-Cited Violations (NCV) of 10 CFR Part 50, Appendix B, Criterion XVI and Technical Specification (TS) 3.4.13 Primary Coolant System (PCS) Operational Leakage for failure to take corrective actions to prevent recurrence of CRDM cracking and leakage, a significant condition adverse to quality which resulted in a violation of TS. Specifically, the licensee failed to include the internal CRDM housing weld build-up area within the scope of corrective actions taken for a 2001 CRDM leakage event and consequently leakage recurred in CRDM-24 in 2012.

Description: In 2001, the licensee discovered a steam leak in the housing of CRDM-21 caused by a through-wall TGSCC at CRDM housing weld No. 3 which was located just below the weld build-up region (weld No. 5). Weld No. 5 consists of a weld material deposit applied to the inside diameter (ID) of the CRDM housing which provides for alignment of the CRDM. This issue was categorized as a significant condition adverse to quality (SCAQ) by the licensee (CPAL0102186) and the licensee's root cause evaluation was documented in RCR/C-PAL-01-02186. The licensee considered this issue a SCAQ based on the procedure EN-LI-102 "Corrective Action Process" definition of "Conditions such as failures, malfunctions, deficiencies, deviations, defective material & equipment, and non-conformances which have resulted in, or could result in, a significant degradation or challenge to nuclear safety." The licensee concluded that the cracks in CRDM-21 were caused by TGSCC which occurred in areas of heavy grinding or machining tool marks. Specifically, this leak was the result of an ID initiated, axially oriented, transgranular crack in the austenitic stainless steel housing material. The failure analysis performed in response to this event identified both axial and circumferential cracks associated with weld No. 3. Extent of condition inspections revealed additional, non-through wall cracks associated with weld No. 3 in 41 of the 44 remaining housings for a total of 42 of 45 housings containing cracks.

In response to the 2001cracking, Palisades replaced all 45 CRDM housings with housings thought to be more resistant to cracking. Principle changes included:

- Elimination of weld No. 2,
- Relocation of weld No. 3 to a higher location thereby minimizing the deposition of crud in the gap between the weld and the bottom plate of the rack and pinion assembly,
- Reduction in residual stresses and cold work on welds by requiring better surface finishes, and
- Use of heat sink welding to reduce ID residual tensile stresses.

Licensee corrective actions taken in response to the 2001 event were limited to butt welds. The inspectors reviewed the licensee actions to determine if they had been sufficient to eliminate one of the 3 necessary factors to cause TGSCC on the CRDM housings: (1) a susceptible material, (2) a corrosive environment and (3) tensile stress. The inspectors identified that the licensee had failed to eliminate one or more of the necessary factors at weld No. 5 (which was not a butt weld) to preclude TGSCC in the replacement housing. Specifically:

- The licensee's 2001 root cause report documented that weld No. 5 is exposed to essentially the same environment as the weld that experienced the cracking (corrosive environment remained unchanged).

- No analysis was completed on the stress conditions for weld No. 5 prior to approving the modified replacement housing design (the potential for residual tensile weld stresses on ID of CRDM surface was not ruled out by analysis and therefore, should have been considered).

- Fabrication restrictions to prohibit grinding were not applied to weld No. 5 (grinding promotes residual tensile stress state on ID of CRDM surface)

- Machining was performed on weld No. 5 during the fabrication process in order to achieve the dimensions and geometry specified in the design. This process induced cold work stresses in the weld.

- Material was changed from type 347 to type 316 stainless steel (both materials are essentially equally susceptible to TGSCC).

Comment [SO1]: Clearly indicate which of these is a corrective action from the 2001 RCE. Are some of these "new" issues that were not identified?

In January of 2002, an NRC special inspection team (SIT) (reference IR 50-2555/01-15) reviewed the licensee proposed corrective actions associated with the through-wall leakage of the CRDM-21 housing caused by TGSCC. The 2001 root cause report reviewed by the NRC stated the action to prevent recurrence was to "develop and implement an inspection plan to address areas and components identified in Attachment C-Extent of Condition". One of the components included in Attachment C was the CRD Mechanism. The ~~recommended~~ licensee's planned action was to perform volumetric inspection of the welds contained in the CRD Mechanism. The table also refers to a susceptibility analysis (EA-C-PAL-01-2186-02 "CRD Upper Housing and Nozzle Weld Susceptibility Comparison") to identify how degradation can be identified in this component. The objective of this document was to provide justification as to why the first weld (weld No. 1) above the reactor head is deemed to be less susceptible than the upper housing welds to failure by TGSCC and should not be included in the extent of condition. The susceptibility analysis excludes weld No. 5 because it is a weld overlay and not a butt weld and was deemed to be less susceptible to TGSCC than the butt welds. By not including weld No. 5 in the susceptibility analysis the licensee did not evaluate the stresses, material and environment of this weld to conclude it is not susceptible to TGSCC. An attachment to this analysis states machining marks were present on weld No. 5 which was identified as a key contributor to the cracking identified in weld No. 3. After this analysis was complete the licensee decided to replace all CRDM housings with the new design and control the fabrication process on the butt welds and the inspection plan would consist of the required ASME inspections. Weld No. 5 was excluded from these corrective actions and no fabrication controls were placed on weld No. 5 to reduce the stresses in this location.

Comment [SO2]: Can we simply state that the licensee subsequently limited the corrective actions and excluded weld 5? Had the licensee included weld five, the corrective actions would have provided reasonable assurance?

On August 12, 2012, Palisades Nuclear Power Station was shut down to investigate an increase in unidentified leakage. During a walk-down performed post shutdown, the licensee discovered the source of the leakage to be a pressure boundary leak from CRDM-24. After further testing, the licensee determined the leak occurred because of a through-wall crack adjacent to weld No. 5.

The licensee formed a root cause team (RCT) staffed with licensee personnel and augmented with input from vendors. The root cause investigation was conducted in accordance with site procedure EN-LI-118 "Root Cause Evaluation Process" and was documented in root cause analysis report CR-PLP-2012-05623. In this report, the licensee's RCT determined that the probable cause of the cracking was:

> "Stresses in the weld build up area due to manufacturing irregularities and misalignments between CRDM-24 upper housing, support tube, and the associated reactor head penetration/CRDM nozzle. Based on lack of cracking found in the other 8 upper housings tested, the failed CRDM-24 upper housing contains an as-yet unidentified additional stress."

The RCT also identified the following contributing cause:

> "Transgranular Stress Corrosion Cracking (TGSCC) initiating within the internal weld build-up material of CRDM-24. The through wall crack initiated in the weld material and then propagated through the base metal until a leak developed in the outer diameter (OD) witness band region at the base of the ID weld build up.

This conclusion was based upon destructive and non destructive examinations (NDE) completed on a section of the failed housing which included the through-wall flaw. The RCT also relied upon vendor technical reports assessing the results of the NDE as well as vendor calculations related to the stresses in the CRDM housings.

To determine the extent of condition, the licensee performed ultrasonic (UT) examinations of weld No. 5 on eight additional CRDM housings. The licensee selected these housings based on being in a similar location on the head as CRDM-24, and previous cracking having been identified in some of these housings prior to the replacement of the CRDM upper housings and seal housings in 2002. The inspectors concluded that this was an adequate sample for an initial extent of condition review based upon the concept that, in light of eight negative exams, the statistical probability of a flaw in the remaining CRDM housings was very low. Additionally, the licensee planned to conduct examinations of more housings during the next refueling outage.

Based upon the recurrence of through-wall leakage in the CRDM housings caused by TGSCC, the inspectors concluded that the licensee actions were not adequate because the appropriate actions to preclude recurrence were within the licensee's ability to foresee and implement. Specifically, the inspectors concluded that the licensee did not effectively implement corrective actions for the 2001 CRDM housing leak resulting in the 2012 CRDM-24 housing leak. Also, in 1991, the Fort Calhoun plant had experienced through-wall leakage due to TGSCC at weld No. 5 of their CRDM housings (same housing design) and this operational experience had been reviewed by the licensee and dismissed. In the licensee's 2001 root cause evaluation, the licensee reviewed the weld build-up region failure by TGSCC at Fort Calhoun in the spare housing and concluded it would not occur at Palisades. This conclusion was based on the assumption that a higher oxygen environment (more aggressive environment) would exist in the spare Fort Calhoun housings than in the inservice Palisades housings. However the licensee did not confirm this assumption, nor did the licensee perform additional testing to determine if the environment of their inservice housings was sufficiently benign to prevent TGSCC. The licensee's 2012 RCT reached a similar conclusion and documented that due to organizational/ programmatic weakness at Palisades, the 1991 Fort Calhoun operating experience was not adequately utilized to include inspection of the weld No. 5. The inspectors identified that the licensee had missed a key opportunity to implement effective corrective actions that could have prevented recurrence of the 2001 leakage event and elected not to pursue. Specifically, in EA-EAR-2001-0426-01 the licensee considered fabricating the replacement housings with Inconel 600 material because it was much more resistant to TGSCC, but ultimately decided not to do so. Additionally, various vendor reports were generated related to this issue. Those reports documented the potential susceptibility of weld No. 5 to TGSCC due to their review of the CRDM housing conditions and available operating experience. The issuance of these documents represents another opportunity for the licensee to identify the susceptibility of weld No. 5 to TGSCC prior to the cracking in CRDM-24.

During the 2012 NRC special inspection, the NRC identified an unresolved item for the Technical Specification pressure boundary leak. The licensee determined the CRDM-24 leakage commenced on July 14, 2012 and the plant continued to operate in this condition, which was contrary to the TS 3.4.13 requirement of limiting PCS operational leakage to no pressure boundary leakage. Based on the review discussed above, unresolved items 05000255/2012012-01 "TS for PCS Pressure Boundary Leakage" and 05000255/2012012-03 "Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality" are closed.

Analysis: The inspectors determined that the licensee's failure to prevent recurrence of TGSCC of the CRDM housings (a significant condition adverse to quality) that resulted in a violation of TS was a performance deficiency that warranted a significance evaluation. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. Specifically the licensee did not limit the likelihood of events that upset plant stability by not taking adequate corrective actions to prevent recurrence of leakage in CRDM housings which represents a pressure boundary leakage. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a component rupture.

Despite the advanced age of the licensees decisions associated with this finding, the inspectors concluded that the finding was indicative of current performance. Specifically, the licensee more recently exhibited similar non-conservative decision making with respect to addressing the potential for CRDM housing cracking and leakage (Section

4OA2.3 (b.2) of this report) and resulting in another finding. However, given that both findings reflect upon the licensee's approach to basically the same equipment and technical issues, the inspectors did not apply a separate cross cutting aspect to this finding in that it is already captured through the other finding.

Enforcement: The inspectors identified NCVs of 10 CFR, Appendix B, Criterion XVI "Corrective Actions", and Technical Specification 3.4.13 "Primary Coolant System Operational Leakage", having a very low safety significance (Green), for failure to prevent the recurrence of leakage in CRDM housings due to TGSCC resulting in the operation of the reactor with pressure boundary leakage, a condition prohibited by TS. Given that both violations relate to the same performance deficiency, they are considered as one finding.

10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that, for significant conditions adverse to quality, the cause of the condition is determined and corrective action taken to preclude repetition.

Technical Specifications 3.4.13 requires PCS operational leakage be limited to "No pressure boundary LEAKAGE" when in Modes 1 through 4.

Contrary to the above, as of August 12, 2012, the licensee had failed to take corrective actions to preclude repetition for a significant condition adverse to quality. Specifically, June 21, 2001, the licensee discovered a through wall leak in CRDM-21 due to TGSCC and failed to reasonably include weld No. 5 in the corrective actions which resulted in a subsequent through wall leak in CRDM-24. The pressure boundary leakage at CRDM-24 began on July 14, 2012, and the plant continued to operate until August 12, 2012, which is contrary to the TS requirement of limiting operational leakage to no pressure boundary leakage.

As a result of the second through wall leak, the licensee took corrective actions which included the development of an inspection plan that would inspect weld No. 5 every outage until all CRDM housings were inspected.

Because these violations were of very low safety significance and were entered into the licensee's corrective action program as CR-PLP-2013-01134, these violations are being treated as an NCVs, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000255/2013002-*XX*; Failure to Prevent Recurrence of a Significant Condition Adverse to Quality).

Comment [SO3]: Is this one violation or two? If two, separate the requirements and the contrary statements.

.2 <u>Failure to Adequately Address the Generic Implications of the Cracking identified in CRDM 24</u>

Introduction: The inspectors identified a Green Finding with an associated Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion V, for failure to follow the root cause procedure. Specifically, the licensee failed to adequately evaluate and document the generic implications of the cause of the cracking identified in CRDM-24.

Description: As a result of the cracking identified in CRDM-24, which was characterized as a SCAQ, the licensee performed a root cause evaluation in accordance with procedure EN-LI-118, "Root Cause Evaluation". This procedure is identified as quality related and serves to implement a portion of the licensee's quality assurance program. While reviewing the 2012 root cause report CR-PLP-2013-05623 related to the cracking identified in CRDM-24, generated as a result of the root cause evaluation, the inspectors identified that the licensee had not appropriately considered the generic implications of the cracking in the extent of condition review. The licensee's proposed corrective actions narrowly focused on weld No. 5, instead of also including broader actions to ensure other CRDM housing welds were fit for their intended service life.

Comment [SO4]: Are these new corrective actions? If so, clearly state what they are.

On March 13, 2013, the inspectors requested that the licensee provide the bases for excluding other CRDM housing welds (weld No. 3 below weld No. 5 and weld No. 4 above weld No. 5) from the scope of planned corrective actions. On March 29, the licensee provided additional information to justify excluding these welds from the scope of the corrective actions. The licensee credited the corrective actions associated with the modifications to the CRDM housing design completed in 2001 as the basis to exclude housing welds No. 3 and No. 4 from additional actions to identify the extent of TGSCC. The corrective actions taken in 2001 included performing heat sink welding, which is a methodology used to reduce the stresses on the inner ID of the weld. The licensee also changed the design to reduce design stresses at weld No. 3 and specified a smoother surface finish (RMS 125) to reduce potential crack initiation points. The licensee stated that these actions would produce compressive stresses on the ID of welds No. 3 and No. 4 making them immune from cracking. The inspectors acknowledged that these actions would reduce the tensile stress at the ID surface and thus reduce the probability of initiating TGSCC.

However, the information provided did not demonstrate that TGSCC would not occur because it did not demonstrate that tensile stress would be eliminated at the ID surface during operation. In particular, repairs completed at the inner surface of weld No. 4, would result in high residual tensile stress at the inside surface of the weld which would promote the initiation of TGSCC. Repairs were also performed on weld No. 3 from the OD surface of the weld. The licensee believed that the last pass heat sink welding process would be sufficient to ensure residual compressive stress would remain at the ID surface of Weld No. 3 even with repairs to the OD surface. However, the licensee had not completed detailed residual weld stress testing or modeling to confirm this assumption.

Comment [SO5]: These actions appear to reflect back to the previous corrective actions and their effectiveness. I do not see this applicable to this issue.

The inspectors identified that the three factors required for TGSCC could still be present at welds No. 3 and No. 4 as follows:

- Corrosive environment – Weld No. 3 would operate in a similar environment as weld No. 5 of the CRDM housing. Weld No. 4 would be exposed to a lower operating temperature than weld No. 5, however, TGSCC can still occur at 250 degrees Fahrenheit as evidenced by the Palisades previous operating experience with cracking identified in the seal housings that operate at even lower temperatures.

- Susceptible material – Welds No. 3 and No. 4 are composed of the same weld filler and base metal materials as weld No. 5 (e.g. weld filler material consistent with the type 316 stainless housing base metal). This material would be equally susceptible to TGSCC, as the type 347 stainless steel and weld filler materials used in the pre-2001 CRDM housing design that developed a through wall leak caused by TGSCC at weld No.3.

- Tensile stresses - While it is assumed that the corrective actions taken in response to the 2001 leak will reduce the potential for tensile stresses to exist on the inner surface of CRDM housings at welds No. 3 and No. 4, especially in light of the repairs made to welds No. 3 and No. 4, it had not been conclusively demonstrated that these tensile stresses have been eliminated. As such, it was not reasonable to conclude that tensile stresses were not present and, therefore, the potential for transgranular stress corrosion cracking had been eliminated.

Comment [SO6]: How does this correlate to the New corrective actions. This discussion seems academic.

Although the root cause report discussed manufacturing irregularities and misalignment between CRDM-24 and the support tube, seismic supports, and the associated reactor head penetration/CRDM nozzle as potential source of stresses leading to cracking, the root cause report also stated that "based on the lack of cracking found in the other 8 upper housings tested, the failed CRDM-24 upper housing contains an as-yet unidentified additional stress." Because the cause of the additional stress was not identified, the licensee had not established a sufficient basis in the RCR to exclude welds No. 3 and No. 4 from the extent of condition review (e.g. potential generic implications) This unknown additional stress as well as the propagation rate represent key differences as related to the cracking identified in 2001. The RCR documents the changes made to the CRDM housings in 2001 to reduce tensile stresses, but it does not document a justification for excluding welds No. 3 and No.4 from an evaluation for generic implications or corrective actions based on the results of the current root cause evaluation.

The inspectors identified that the licensee had not followed Procedure EN-LI-118 "Root Cause Evaluation," in the root cause review of the CRDM-24 leak as documented in report CR-PLP-2013-05623. Section 5.5 (12)e of EN-LI-118 required that the licensee "perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affects other SSC's." Additional details are provided in the procedure on how to conduct and document the evaluation. In this case, the inspectors identified that the licensee had not addressed or documented a basis in RCR CR-PLP-2013-05623 to exclude welds No. 3 and No. 4 from the generic factors discussed above that led to the 2012 leak in CRDM-24 (e.g. TGSCC at weld No. 5) sufficiently to meet the intent of the procedural requirements. The licensee entered this issue into the corrective action program as CR-PLP-2013-01500. To restore compliance with the procedure, the licensee intended to revise the inspection plan to add additional corrective actions to inspect a sample of welds No. 3 and No. 4 for TGSCC during the upcoming refueling outage.

Analysis: The inspectors determined that the failure to adequately evaluate and document the generic implications of the cause of the cracking identified in CRDM-24 in accordance with the root cause procedure EN-LI-118 was a performance deficiency that warranted a significance evaluation. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. The inspectors also answered "yes" to the More-than-Minor screening question "if left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, absent NRC identification, the licensee would not have completed further evaluations or inspections of CRDM housing welds which could have resulted in additional CRDM housing failure and leakage by TGSCC. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609, Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a large component rupture.

The inspectors determined that the primary cause of the failure to adequately consider welds No. 3 and No. 4 in the generic implications section of the root cause report related to the cross-cutting component of Human Performance, Decision Making, because licensee staff did not use conservative assumptions in decision making. Specifically, the licensee did not use conservative assumptions when excluding welds No. 3 and No. 4 as being susceptible to TGSCC and therefore include them in the generic implications section of the root cause report. (Item H.1(b) of IMC 310).

<u>Enforcement:</u> The inspectors identified a NCV of 10 CFR, Appendix B, Criterion V "Instructions, Procedures and Drawings", having a very low safety significance (Green), for failure to adequately evaluate and document the generic implications of the cause of cracking identified in CRDM-24 as it relates to welds No. 3 and No. 4 in accordance with the root cause procedure.

Title 10 CFR, Appendix B, Criterion V "Instructions, Procedures and Drawings requires in part, "Activities affecting quality shall be prescribed by documented procedures and shall be accomplished in accordance with these procedures."

Procedure EN-LI-118 "Root Cause Evaluation Process," Revision 17 states:

- 5.5 (12)e: perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affects other SSCs , organizations or work processes. Use the two step process in accordance with attachment 9.7
- Attachment 9.7: Determine whether the occurrence/consequence (problem) is isolated, or whether it has broader (generic or common mode) implications. Achieve this by asking the following questions:
    i. Could this happen to equipment that is similar in function, design, or service condition?
    ii. Could this happen to a group of components? (components of the same construction or materials that could be similarly affected by one condition)

- Attachment 9.7: Document the results of the above considerations. Include the following items in the write up:
  i. Generic Implications (Is this problem/ cause limited to this component/equipment, or does it apply to others as well)
  ii. Existing broader (generic/common mode) considerations
- 5.5(15)(10)c&f: Document proposed corrective actions and due dates to address valid generic implications. If no corrective action is recommended for a valid generic implication then document the basis for this conclusion and any risk or consequence identified as a result of taking no action.

Contrary to the above, from February 24, 2013 through April 18, 2013, the licensee failed to ~~perform~~ accomplish ~~an~~ activitiesy affecting quality in accordance with procedure EN-LI-118, which was being implemented to correct a significant condition adverse to quality. Specifically, the licensee did not evaluate and document the existing broader (generic/common mode) considerations associated with TGSCC at CRDM housing welds No. 3 and No. 4, including . . . ~~Consequently, the licensee failed to propose corrective actions for the generic implications of TGSCC at CRDM housing welds No. 3 and No. 4 or to provide reasonable rationale why corrective actions were unnecessary. The licensee was considering adding welds No. 3 and No. 4 into its inspection plan for activities to be performed during the next refueling outage.~~ Because of the very low safety significance and because the licensee entered this issue into their corrective action program (CR-PLP-2013-01500), it is being treated as a NCV consistent with Section 2.3.2 of the Enforcement Policy (NCV 05000255/2013003-xx).

Comment [SO7]: Include a representative indication of what NEW corrective actions should have been considered.

4OA5 Other Activities

.1 (Closed) Unresolved Item 05000255/2012012-02: Potential Inadequate Degradation Evaluation of CRDM Housings

During a Special Inspection performed in August 2012, NRC inspectors identified an issue which could not be resolved without additional information (Unresolved Issue (URI)). This issue was associated with the rate of growth of the crack which created the through wall leak in CRDM-24, discovered on August 12, 2012. Identification of this crack growth rate is significant in determining appropriate intervals for future inspections to provide reasonable assurance that CRDM housing leakage will not recur.

Preliminary failure analysis data available at the time of the inspection indicated that the observed cracking was due to TGSCC. Cracking of this type is normally due to the presence of oxygen and chlorides at the location of the crack. When examining the fracture surface at the location the through-wall leak occurred, the licensee identified six concentric rings (beach marks) propagating in a radial direction from the ID out towards the OD of the housing. Beach marks are normally associated with fatigue failures and indicate the number of stress cycles from crack initiation to crack failure. In this case, there was no evidence that fatigue contributed to the failure. Despite the lack of evidence of fatigue, it was apparent that the crack which resulted in the CRDM-24 leak grew in increments. It was not, however, immediately apparent whether the increments were related to oxygen ingress (refueling outages) or temperature/pressure cycles (heatups/cooldowns).

At the time of the original inspection, 5 time intervals for through wall crack growth were under consideration. Two were based on literature crack growth data and three were based on interpretations of the beach marks. These time intervals were:

- Based on literature data, one contractor estimated that a 10% through wall flaw would require 4 years to reach 50% through wall.
- Based on literature data another contractor estimated the crack growth rate to be 2.1 x 10^{-5} in/hr or 0.18 in/yr. This is approximately three times faster than the crack growth rate proposed in the above mentioned rate.
- Based on the concept of oxygen ingress at refueling outages 6 cycles of 18 months duration would require 9 years for the crack to grow through wall
- Based on the concept of temperature/pressure cycles, the plant experienced 6 cold shutdowns in approximately 2 years preceding the crack. This equates to 2 years for the crack to grow through wall.
- Based on the concept that oxygen is required for crack growth and that oxygen is rapidly purged from the CRDM housings due to leakage past the seals, crack growth occurs only during the first few weeks of operation following a refueling outage, followed by no growth for the remaining period of operation when oxygen concentrations are low. This equates to 6 oxygen ingress events (irrespective of time between events) for the crack to grow through wall.

NRC inspectors including technical experts from NRC Headquarters performed a follow-up inspection to determine if the assumptions made by the licensee were conservative and the planned actions bounded those conservative assumptions. The inspectors reviewed a variety of documents associated with crack growth and inspection intervals. The inspectors noted the following statements included in the root cause report and vendor documents related to the determination of the appropriate crack growth rate:

- The laboratory conducting the failure analysis concluded, it could not be conclusively determined if the beach marks corresponded to refueling outages, (i.e., 18 month cycle) or shorter periods as occurred during outages over the past 24 months
- Palisades CRDM-21 leaked at weld No. 3 in 2001. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 failure. In calculating the crack growth rate of this crack, one contractor utilized an interval between beach marks which is much shorter than refueling outages. The intervals used are consistent with plant thermal cycles in which oxygen may or may not have been admitted into the CRDMs.
- A spare CRDM housing at Ft Calhoun leaked at weld No. 5 in 1990. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 Palisades failure. In calculating the crack growth rate of this crack, Ft Calhoun stated that the beach marks were related to refueling cycles. Ft Calhoun also performed calculations indicating that the oxygen level at the location of the flaw did not change with time (including in response to refueling outages) because the spare CRDM housing was not vented. Ft Calhoun's evaluation indicated that oxygen levels at the vicinity of the crack would have begun to decline through diffusion and convection had the intervals between outages been much longer than 18 months. This is interpreted to mean that the beach marks at Ft Calhoun are in response to pressure/thermal cycles.

- In at least one instance Palisades needed to repair the seals on a reactor coolant pump at a time other than an outage. This necessitates draining some of the water from the reactor coolant system and venting (admitting oxygen into) the CRDM housing. This represents an additional oxygen ingress event not included when determination of time to cracking is based on refueling outages.
- In its inspection plan, Palisades states that it will inspect all CRDM housings over the next 4 refueling outages, i.e., the interval between inspections is 1 refueling outage

Based on the above review, the inspection team noted that there were certain non conservative statements contained in the Root Cause Report and the inspection plan. These included:

- The crack growth rate based on refueling outages was understated. If oxygen ingress is related to beach marks, given the oxygen ingress event which occurred to repair reactor coolant pump seals, six beach marks would occur in a maximum of five refueling intervals rather than the six refueling intervals that were used to calculate the crack growth rate in the root cause report.
- The crack growth rate based on heat up and cool down cycles is overstated. The value in the root cause is based on 11 months. While 6 shutdowns did occur at the plant in 11 months several of these events did not result in pressure/temperature changes of the reactor coolant system. The appropriate time frame is 24 months rather than 11.
- The inspection plan contains a non conservative statement: "However, once the crack has been initiated it propagates over 4 to 5 operating cycles prior to going through wall." While this statement does reflect one of the proposed theories for crack growth, sufficient evidence to demonstrate reasonable assurance that this theory is correct, and thereby overcome the non-conservatism of this statement, was not provided.

Despite the existence of the non conservatisms stated above, the inspectors concluded:

- Sufficient evidence to conclusively determine the rate of crack growth does not exist.
- Crack growth based on pressure/temperature cycles is the most conservative of the potential crack growth mechanisms. In the absence of reasonable assurance of the correctness of less conservative mechanisms, through wall crack growth in two years must be utilized for regulatory purposes.
- The licensee has not formally committed to any of the crack growth mechanisms discussed.
- The licensee's inspection program includes inspection of all of the CRDM housings over the next 4 refueling outages. Approximately 25% of the housings will be inspected during each outage. The inspection of 25% of the CRDM housings each interval is sufficient to indicate that, in the event no indications are found during a given inspection, that the probability that flaws exist in other housings is extremely low. As such, it may be considered that the inspection of approximately 25% of the CRDM housings every refueling outage bounds all the crack growth rate mechanisms considered.

The inspectors considered this approach to inspection to be both acceptable and sufficient justification to close this URI.

4OA6 Management Meetings

.2 Interim Exit Meetings

An interim exit was conducted for:

- The results of the selected issue follow-up inspection, with Mr. T. Vitali, Site Vice President on April 18, 2013.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

B. Davis, Engineering Director
O. Gustafson, Licensing Manager
T. Foudy, Engineering Supervisor
B. Williams, Engineer
B. Dotson, Licensing

LIST OF ITEMS OPENED, CLOSED, DISCUSSED

Closed

05000255/2012012-01	URI	TS for PCS Pressure Boundary Leakage
05000255/2012012-02	URI	Potential Inadequate Degradation Evaluation of CRDM Housings
05000255/2012012-03	URI	Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality

Opened and Discussed

None.

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

4OA5 <u>Other Activities</u>

LIST OF ACRONYMS USED

RELEASE IN ENTIRETY

Giessner, John

From: Giessner, John
Sent: Tuesday, April 30, 2013 4:16 PM
To: Sanchez Santiago, Elba
Subject: LCO vs action

After reading the contradictory document call the enf manual TS citing, I could support what Dave wants. As long as we discuss what they did before (that paragraph I added). The manual says you can cite against the LCO(p338) in certain cases (but there is no violation, p336 says you have to have both LCO not met and time not met).

Anyway I wanted to say I moved to a more neutral position – after thinking I should be more facilitative.

I'd prefer my/your approach
j

C 3.5

Hills, David

From: Sanchez Santiago, Elba
Sent: Thursday, May 02, 2013 3:38 PM
To: Hills, David
Subject: RE: Palisades Violation Approach

Dave,

I think what Steve is currently looking for is what makes the 2012 incident different from 2001 to ensure we aren't hitting them twice for the same thing. I included some sentences on that point in the report. If that doesn't work, we would likely need to open a URI to approach the issue from the topical report requirements standpoint (I'm not sure we'd be able to complete it in time for the current report).

-Elba

From: Hills, David
Sent: Thursday, May 02, 2013 3:34 PM
To: Sanchez Santiago, Elba
Subject: Palisades Violation Approach

Elba,

Just a thought. Another possible approach to better satisfy Steve. Takes your idea about Criterion II, but builds upon it via the licensee's QA program.

10 CFR 50, Appendix B, Criterion II requires that the quality assurance program be documented by written policies, procedures, or instructions and shall be carried out throughout plant life in accordance with these policies, procedures, or instructions.

The licensee's quality assurance topical report states that for significant conditions adverse to quality, the program provides for cause evaluation.

Procedure EN-LI-118 Root Cause Evaluation Process prescribes the licensee's quality assurance controls for cause evaluation.

Procedure EN-LI-118 states . . . . . .

Do you think the QA Topical Report is the link Steve is looking for? Or do you think he is looking for something more? Of course, you would actually need to verify what their QA topical report actually says, but the above is typical. We typically don't reference licensee's QA programs anymore in violations, but really nothing wrong with doing so if it is needed. We used to do it frequently in the 80s.

- Dave

Sanchez Santiago, Elba

From:	Sanchez Santiago, Elba
Sent:	Friday, May 03, 2013 3:17 PM
To:	Hills, David
Cc:	Holmberg, Mel
Subject:	Palisades Report
Attachments:	Palisades Input to DRP Report 2013 002 URI EMS.docx
Importance:	High

Dave,

Attached is the latest revision of the Palisades input for your review.

-Elba

UNITED STATES
NUCLEAR REGULATORY COMMISSION
LISLE, IL 60532-4352

May XX, 2012

MEMORANDUM TO: Thomas Taylor
Senior Resident Inspector
Palisades Nuclear Plant

FROM: David Hills, Chief
Engineering Branch 3
Division of Reactor Safety

SUBJECT: PALISADES NUCLEAR PLANT
DRS INPUT TO INTEGRATED REPORT 05000255/2013002

Enclosed is the report input for the Palisades Nuclear Plant, Inspection Report 05000255/2013002. This report input documents completion of our review of Unresolved Items 05000255/2012012-01, "TS for PCS Pressure Boundary Leakage," 05000255/2012012-02, "Potential Inadequate Degradation Evaluation of CRDM Housings," and 05000255/2012012-03, "Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality." This report also completes one sample of the Problem Identification and Resolution, Selected Issue Follow-up inspection in accordance with IP 71152. I have reviewed this input to confirm compliance with Inspection Manual Chapter (IMC) 0612 and IMC 0305. This input is ready for inclusion into the integrated report and dissemination to the public.

Please input the following post Inspection Data into RPS:

Inspection Procedure	**Procedure Status** – see below: Incomplete, Complete, Complete by reference, Complete-full sample not available, Complete – opportunity to apply procedure not available, Not Applicable.	**Sample Size** – As documented in Scope Section If less than full sample size documented in the report input, the inspector must provide a justification below to enter into RPS and support the procedure status selected
71152	Complete	1

Inspection Report Item and Type (AV, FIN, NCV, URI or VIO)	**Cornerstone** (IE, MS, BI, EP, OR, PR, MISC)	**Cross Cutting Aspect** (H.n(i), P.n(i), S.n(i))	**Responsible Person/Owner**	**Procedure or TI** (71111.07T)	**RPS Branch Code (e.g. closeout responsibility)** EB1 3820 EB2 3870 EB3 3840 PST (RP) 3860 PSB (Safeguards) 3850 OB 3810
NCV-XXX	IE	n/a	E. Sanchez Santiago	71152	3820
NCV-XXX	IE	H.1(b)	E. Sanchez Santiago	71152	3820

Enclosure: Input to Inspection Report 05000255/2013002

cc w/encl: J. Giessner, Chief
C. Hernandez, Site Admin Assistant

CONTACT: E. Sanchez Santiago, DRS
(630) 829-9715

DOCUMENT NAME: G:\DRSIII\DRS\Work in Progress\-Palisades Input to DRP Report 2013 002 URI EMS.docx
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OFFICE	RIII		NRR		RIII		NRR	
NAME	ESanchezSantiago		DAlley		DHills		TLupold	
DATE	5/ /13							

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Cover Letter

X Green findings involving a violation were identified. Include the following:

Based on the results of this inspection, two NRC-identified findings of very low safety significance (Green) were identified. These findings were determined to involve a violation of NRC requirements. However, because of the very low safety significance and because the issues were entered into your corrective action program, the NRC is treating the issue as Non-Cited Violation, in accordance with Section 2.3.2 of the NRC Enforcement Policy.

TITLE PAGE

Inspectors: D. Alley, Senior Materials Engineer
E. Sanchez Santiago, Reactor Inspector

SUMMARY OF FINDINGS

A. NRC-Identified and Self-Revealed Findings

Cornerstones: Initiating Events

- Green. A self-revealing Green Finding with associated Non-Cited Violations (NCV) of 10 CFR Part 50, Appendix B, Criterion XVI and Technical Specification (TS) 3.4.13 Primary Coolant System (PCS) Operational Leakage, was identified for failure to take corrective actions to prevent recurrence of Control Rod Drive Mechanism (CRDM) cracking and leakage, a significant condition adverse to quality (SCAQ), and resulting in operation of the reactor with PCS pressure boundary leakage. Specifically, for Criterion XVI the licensee failed to include the internal CRDM housing weld build-up area within the scope of corrective actions taken for a 2001 CRDM through wall leak on CRDM-21, caused by transgranular stress corrosion cracking (TGSCC). Subsequently, a through wall leak recurred in the weld build-up area on CRDM-24 in 2012 due to TGSCC. As a result, the licensee operated with PCS pressure boundary leakage, which is not allowed by TS 3.4.13. Further, because the licensee was not aware that the leakage was PCS pressure boundary leakage, the licensee did not implement the associated TS action statement. The licensee replaced CRDM-24 upper housing and wrote CR-PLP-2013-01134.

  The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone objective to limit the likelihood of events that upset plant stability. The issue was associated with the attribute of equipment performance. Specifically, the licensee did not take adequate corrective actions to prevent recurrence of leakage in CRDM housings, which represents pressure boundary leakage. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System Loss of Coolant Accident (LOCA) initiator contributor. The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the slow rate of change for leakage for this cracking mechanism and this type of material. Type 316 stainless steel material under TGSCC will experience leakage rates well below a small break LOCA, which would be observed through the crack, alerting operators to take action to shut down the plant prior to experiencing a component rupture. The cause of this finding, non-conservative decision making, occurred over ten years ago and is well outside of the nominal three year period in IMC 0612; and would not be indicative of current performance, unless there were other opportunities to identify the issue; therefore, the inspectors concluded this was not indicative of current performance. However more recently, the licensee exhibited non-conservative decision making with respect to addressing the potential for CRDM housing cracking and leakage during the recent root cause (Section 4OA2.3 (b.2) of this report), resulting in another finding. This cross-cutting aspect will be captured through the other finding. (Section 4OA2.3(b.1))

- Green. The inspectors identified a Finding with an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, for the licensee's failure to accomplish quality activities in accordance with the prescribed procedures. Specifically, the licensee failed to adequately evaluate and document the generic implications of the cause of the cracking identified in CRDM-24 in accordance with root cause procedure EN-LI-118. This issue was entered into the licensee's corrective action program under CR-PLP-2013-01500.

  The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because the inspectors answered "yes" to the More-than-Minor screening question, "if left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, absent NRC identification, the licensee would not have completed further evaluations or inspections of CRDM housing welds, which could have resulted in additional CRDM housing failure and leakage by TGSCC. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System LOCA initiator contributor. The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609, Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the slow rate of change for leakage for this cracking mechanism and this type of material. Type 316 stainless steel material under TGSCC will experience leakage rates well below a small break LOCA, which would be observed through the crack, alerting operators to take action to shut down the plant prior to experiencing a component rupture. The inspectors determined that the primary cause of the failure to adequately consider welds No. 3 and No. 4 in the generic implications section of the root cause report (RCR) related to the cross-cutting component of Human Performance, Decision Making, because licensee staff did not use conservative assumptions in decision making. Specifically, the licensee did not use conservative assumptions when excluding welds No. 3 and No. 4 as being susceptible to TGSCC when there was not enough information to exclude them from consideration. (Item H.1(b)). (Section 4OA2.3(b.2))

B. **Licensee-Identified Violations**

No violations of significance were identified.

REPORT DETAILS

4. REACTOR SAFETY

4OA2 Identification and Resolution of Problems (71152)

.3 Selected Issue Follow-up Inspection: Through Wall Leakage of CRDM-24, (This inspection is part of the additional inspections included in the Palisades Deviation letter)

a. Inspection Scope

On August 12, 2012, the licensee shut down the plant to investigate an increase in unidentified leakage. The source of the leakage was determined to be a crack in CRDM-24. The NRC dispatched a special inspection team (SIT) to review the CRDM-24 leakage event. The results of that inspection are provided in Inspection Report 05000255/2012012. The licensee completed an evaluation to determine the cause of the cracking (CR-PLP-2012-05623).

From March 4, 2013 to March 15, 2013, the inspectors completed one inspection sample regarding problem identification and resolution based upon review of the licensee's RCR contained in corrective action document CR-PLP-2012-05623. In addition, the inspectors performed reviews related to three Unresolved Items (URI) identified during the SIT inspection:

- URI 05000255/2012012-01 TS for PCS Pressure Boundary Leakage. (The closure of this URI is documented in section 4OA2.3 (b.1) of this report.)
- URI 05000255/2012012-02 Potential Inadequate Degradation Evaluation of CRDM Housings (The closure of this URI is documented in section 4OA5.1 of this report)
- URI 05000255/2012012-03 Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality (The closure of this URI is documented in section 4OA2.3 (b.1) of this report.)

The inspectors reviewed the licensee's actions in accordance with performance attributes identified in IP 71152. Specifically, the inspectors reviewed licensee corrective action records to determine if: (1) the problems were accurately identified; (2) operability and reportability were adequately ascertained; (3) extent of condition and generic implications were appropriately addressed; (4) classification and prioritization of the problem were commensurate with safety significance; (5) root and contributing causes were identified; (6) corrective actions were appropriately focused to correct the problem; and (7) timely corrective actions were completed or proposed commensurate with the safety significance of the issues.

b. Findings

.1 Failure to Take Corrective Actions to Prevent Recurrence of CRDM Housing Cracking and Leakage

Introduction: A self-revealing Green Finding with associated NCV of 10 CFR Part 50, Appendix B, Criterion XVI and TS 3.4.13 PCS Operational Leakage, was identified for failure to take corrective actions to prevent recurrence of CRDM cracking and leakage, a SCAQ, and resulting in operation of the reactor with PCS pressure boundary leakage. Specifically, for Criterion XVI the licensee failed to include the internal CRDM housing weld build-up area within the scope of corrective actions taken for a 2001 CRDM through wall leak on CRDM-21 caused by TGSCC. Subsequently, a through wall leak recurred in the weld build-up area on CRDM-24 in 2012 due to TGSCC. As a result, the licensee operated with PCS pressure boundary leakage, which is not allowed by TS 3.4.13. Further, because the licensee was not aware that the leakage was PCS pressure boundary leakage, the licensee did not implement the associated TS action statement.

Description: In 2001, the licensee discovered a steam leak in the housing of CRDM-21 caused by a through-wall TGSCC at CRDM housing weld No. 3, which was located just below the weld build-up region (weld No. 5). Weld No. 5 consists of a weld material deposit applied to the inside diameter (ID) of the CRDM housing which provides for alignment of the CRDM. This issue was categorized as a SCAQ by the licensee (CPAL0102186) because it represented a break in the reactor system pressure boundary. The licensee's root cause evaluation was documented in RCR/C-PAL-01-02186 and concluded that the cracks in CRDM-21 were caused by TGSCC, which occurred in areas of heavy grinding or machining tool marks. Specifically, this leak was the result of an ID initiated, axially oriented, transgranular crack in the austenitic stainless steel housing material. The failure analysis performed in response to this event identified both axial and circumferential cracks associated with weld No. 3. Extent of condition inspections revealed additional, non-through wall cracks associated with weld No. 3 in 41 of the 44 remaining housings for a total of 42 of 45 housings containing cracks.

In response to the 2001 cracking, Palisades replaced all 45 CRDM housings with housings thought to be more resistant to cracking. Principle changes included:

- Elimination of weld No. 2,
- Relocation of weld No. 3 to a higher location thereby minimizing the deposition of crud in the gap between the weld and the bottom plate of the rack and pinion assembly,
- Reduction in residual stresses and cold work on welds by requiring better surface finishes, and
- Use of heat sink welding to reduce ID residual tensile stresses.

In January of 2002, an NRC SIT (reference IR 50-2555/01-15) reviewed the licensee proposed corrective actions associated with the through-wall leakage of the CRDM-21 housing caused by TGSCC. The 2001 RCR reviewed by the NRC stated the action to prevent recurrence was to "develop and implement an inspection plan to address areas and components identified in Attachment C-Extent of Condition. One of the components included in Attachment C was the CRDM. The recommended action was to perform volumetric inspection of the welds contained in the CRDM. Subsequently, the licensee changed the corrective actions and excluded weld No. 5.

Following the subsequent 2012 CRDM-24 leak, the licensee determined the leak occurred because of a through-wall crack adjacent to weld No. 5. The licensee formed a root cause team (RCT) staffed with licensee personnel and augmented with input from vendors. The root cause investigation was conducted in accordance with site procedure EN-LI-118 "Root Cause Evaluation Process" and was documented in root cause analysis report CR-PLP-2012-05623. In this report, the licensee's RCT determined that the probable cause of the cracking was:

> "Stresses in the weld build up area due to manufacturing irregularities and misalignments between CRDM-24 upper housing, support tube, and the associated reactor head penetration/CRDM nozzle. Based on lack of cracking found in the other eight upper housings tested, the failed CRDM-24 upper housing contains an as-yet unidentified additional stress."

The RCT also identified the following contributing cause:

> "TGSCC initiating within the internal weld build-up material of CRDM-24. The through wall crack initiated in the weld material and then propagated through the base metal until a leak developed in the outer diameter (OD) witness band region at the base of the ID weld build up.

This conclusion was based upon destructive and non destructive examinations (NDE) completed on a section of the failed housing, which included the through-wall flaw. The RCT also relied upon vendor technical reports assessing the results of the NDE as well as vendor calculations related to the stresses in the CRDM housings.

To determine the extent of condition, the licensee performed ultrasonic (UT) examinations of weld No. 5 on eight additional CRDM housings. The licensee selected these housings based on being in a similar location on the head as CRDM-24, and previous cracking having been identified in some of these housings prior to the replacement of the CRDM upper housings and seal housings in 2002. The inspectors concluded that this was an adequate sample for an initial extent of condition review based upon the concept that, in light of eight negative exams, the statistical probability of a flaw in the remaining CRDM housings was very low. Additionally, the licensee planned to conduct examinations of more housings during the next refueling outage.

The inspectors concluded that the licensee actions following the 2001 leak were not adequate because the appropriate actions to preclude recurrence were within the licensee's ability to foresee and implement. Specifically, the inspectors concluded that the licensee did not effectively implement corrective actions for the 2001 CRDM housing leak resulting in the 2012 CRDM-24 housing leak.

Licensee corrective actions taken in response to the 2001 event were limited to butt welds. The inspectors reviewed the licensee actions to determine if they had been sufficient to eliminate one of the three necessary factors to cause TGSCC on the CRDM housings: (1) a susceptible material, (2) a corrosive environment and (3) tensile stress. The inspectors identified that the licensee had failed to eliminate one or more of the necessary factors at weld No. 5 (which was not a butt weld) to preclude TGSCC in the replacement housing. Specifically:

- The licensee's 2001 RCR documented that weld No. 5 is exposed to essentially the same environment as the weld that experienced the cracking (corrosive environment remained unchanged).

- No analysis was completed on the stress conditions for weld No. 5 prior to approving the modified replacement housing design (the potential for residual tensile weld stresses on ID of CRDM surface was not ruled out by analysis and therefore, should have been considered).

- Fabrication restrictions to prohibit grinding were not applied to weld No. 5 (grinding promotes residual tensile stress state on ID of CRDM surface).

- Machining was performed on weld No. 5 during the fabrication process in order to achieve the dimensions and geometry specified in the design. This process induced cold work stresses in the weld.

- Material was changed from type 347 to type 316 stainless steel (both materials are essentially equally susceptible to TGSCC).

Also, in 1991, the Fort Calhoun plant had experienced through-wall leakage due to TGSCC at weld No. 5 of its CRDM housings (same housing design) and this operational experience had been reviewed by the licensee and dismissed. In the licensee's 2001 root cause evaluation, the licensee reviewed the weld build-up region failure by TGSCC at Fort Calhoun and concluded it would not occur at Palisades. This conclusion was based on the assumption that a higher oxygen environment (more aggressive environment) would exist in the Fort Calhoun housings than in the inservice Palisades housings. However the licensee did not confirm this assumption, nor did the licensee perform additional testing to determine if the environment of their inservice housings was sufficiently benign to prevent TGSCC. The licensee's 2012 RCT reached a similar conclusion and documented that due to organizational/ programmatic weakness at Palisades, the 1991 Fort Calhoun operating experience was not adequately utilized to include inspection of the weld No. 5. The inspectors identified that the licensee had missed a key opportunity to implement effective corrective actions that could have prevented recurrence of the 2001 leakage event and elected not to pursue. Specifically, in EA-EAR-2001-0426-01 the licensee considered fabricating the replacement housings with Inconel 600 material because it was much more resistant to TGSCC, but ultimately decided not to do so. Additionally, various vendor reports were generated related to this issue in the mid 2000's. Those reports documented the potential susceptibility of weld No. 5 to TGSCC based upon a review of the CRDM housing conditions and available operating experience. The reports also noted that weld No. 5 was not inspected in any of the housings in 2001. One report in 2003 noted that weld No. 5 should have been examined as part of the action from the 2001 events since it was similar to Fort Calhoun. The issuance of these documents represented another opportunity for the licensee to identify the susceptibility of weld No. 5 to TGSCC prior to the cracking in CRDM-24.

The inspectors concluded the corrective actions taken in response to the 2001 CRDM through wall leak from TGSCC, a SCAQ, were not effective to preclude repetition. In particular, a through wall leak did recur on a CRDM from TGSCC. This issue was within the licensee's ability to foresee and correct; therefore, the issue was a performance deficiency. During the 2012 NRC special inspection, the NRC identified an URI for the TS pressure boundary leak. LCO 3.4.13 does not allow any pressure boundary leakage.

Further, Action B, associated with this LCO, requires shutdown to mode 3 in six hours and mode 5 in 36 hours for such leakage. The licensee determined the CRDM-24 leakage commenced on or around July 14, 2012, and the plant continued to operate in this condition until August 12, 2012. Because the licensee was not aware of the existence of pressure boundary leakage, it failed to shut down the unit in six hours for a pressure boundary leak as required by TS 3.4.13 Action B. The NRC previously assessed the site's action for increasing unidentified leakage as part of the SIT. The NRC determined, at the time of higher unidentified leakage, the site took appropriate actions to attempt to locate the leak, eventually shutting down around .3 gallons per minute (gpm) leakage (earlier than the TS value of 1 gpm value for unidentified leakage). The licensee did not identify the source of the leakage as pressure boundary leakage until the shutdown on August 12, 2012, when a tour near the vessel head revealed the leaking housing. The pressure boundary leakage resulted in a TS violation due to the performance deficiency associated with the above mentioned Criterion XVI violation

Based on the review discussed above, URIs 05000255/2012012-01 "TS for PCS Pressure Boundary Leakage" and 05000255/2012012-03 "Potential Failure to Take Corrective Actions to Prevent Recurrence of a Significant Condition Adverse to Quality" are closed.

Analysis: The inspectors determined that the licensee's failure to prevent recurrence of TGSCC of the CRDM housings (a SCAQ) that resulted in a violation of TS was a performance deficiency that warranted a significance evaluation. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone objective to limit the likelihood of events that upset plant stability. The issue was associated with the attribute of equipment performance. Specifically, the licensee did not take adequate corrective actions to prevent recurrence of leakage in CRDM housings, which represents pressure boundary leakage. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the slow rate of change for leakage for this cracking mechanism and this type of material. Type 316 stainless steel material under TGSCC will experience leakage rates well below a small break LOCA, which would be observed through the crack, alerting operators to take action to shut down the plant prior to experiencing a component rupture.

The cause of this finding, non-conservative decision making, occurred over ten years ago and is well outside of the nominal 3 year period in IMC 0612; and would not be indicative of current performance, unless there were other opportunities to identify the issue; therefore, the inspectors concluded this was not indicative of current performance. However more recently, the licensee exhibited non-conservative decision making with respect to addressing the potential for CRDM housing cracking and leakage during the recent root cause (Section 4OA2.3 (b.2) of this report), resulting in another finding. This cross-cutting aspect will be captured through the other finding.

Enforcement: During this inspection, the inspectors identified two NCVs of NRC requirements:

Title 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that, for significant conditions adverse to quality, the cause of the condition is determined and corrective action taken to preclude repetition.

TS LCO 3.4.13 requires PCS operational leakage be limited to "No pressure boundary LEAKAGE" when in Modes 1 through 4.

Contrary to the above, as of August 12, 2012, the licensee had failed to take corrective actions to preclude repetition for a SCAQ. Specifically, on June 21, 2001, the licensee discovered a through wall leak in CRDM-21 due to TGSCC and failed to reasonably include weld No. 5 in the corrective actions which resulted in a subsequent through wall leak in CRDM-24 due to TGSCC.

Contrary to the above, on or around July 14, 2012, PCS pressure boundary leakage at CRDM-24 existed while in Mode 1. Further, because the licensee was not aware that the leakage was PCS pressure boundary leakage, the licensee did not implement the associated TS action statement.

As a result of the second through wall leak, the licensee took corrective actions, which included the development of an inspection plan that would inspect weld No. 5 every outage until all CRDM housings were inspected.

Because these violations were of very low safety significance and were entered into the licensee's corrective action program as CR-PLP-2013-01134, these violations are being treated as an NCVs, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000255/2013002-*XX*; Failure to Take Corrective Action to Prevent Recurrence of CRDM Pressure Boundary Leakage).

.2 Failure to Adequately Address the Generic Implications of the Cracking identified in CRDM-24

Introduction: The inspectors identified a Finding with an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, for the licensee's failure to accomplish quality activities in accordance with the prescribed procedures. Specifically, the licensee failed to adequately evaluate and document the generic implications of the cause of the cracking identified in CRDM-24 in accordance with root cause procedure EN-LI-118. This issue was entered into the licensee's corrective action program under CR-PLP-2013-05623.

Description: As a result of the cracking identified in CRDM-24, which was characterized as a SCAQ, the licensee performed a root cause evaluation in accordance with procedure EN-LI-118, "Root Cause Evaluation". This procedure was identified as quality related and served to implement control pursuant to the licensee's quality assurance program. While reviewing the 2012 RCR (CR-PLP-2013-05623) related to the cracking identified in CRDM-24, the inspectors identified that the licensee had not appropriately considered the generic implications of the cracking in the extent of condition review. The licensee's proposed corrective actions, as a result of the 2012 RCR, narrowly focused on weld No. 5, instead of also including broader actions to ensure other CRDM housing welds were fit for their intended service life. These corrective actions consist of performing inspections of welds No. 5 on all CRDM housing.

On March 13, 2013, the inspectors requested that the licensee provide the bases for excluding other CRDM housing welds (weld No. 3 below weld No. 5 and weld No. 4 above weld No. 5) from the 2012 RCR scope of planned corrective actions. On March 29, 2013, the licensee provided additional information to justify excluding these welds from the scope of the corrective actions. The licensee credited the corrective actions associated with the modifications to the CRDM housing design completed in 2001 as the basis to exclude housing welds No. 3 and No. 4 from additional actions to identify the extent of TGSCC. The corrective actions taken in 2001 included performing heat sink welding, which is a methodology used to reduce the stresses on the inner ID of the weld. The licensee also changed the design to reduce design stresses at weld No. 3 and specified a smoother surface finish (RMS 125) to reduce potential crack initiation points. The licensee stated that these actions would produce compressive stresses on the ID of welds No. 3 and No. 4 making them immune from cracking. The inspectors acknowledged that these actions would reduce the tensile stress at the ID surface and thus reduce the probability of initiating TGSCC. However, the information provided did not demonstrate that TGSCC would not occur because it did not demonstrate that tensile stress would be eliminated at the ID surface during operation.

The inspectors identified that the three factors required for TGSCC could still be present at welds No. 3 and No. 4 as follows:

- Corrosive environment – Weld No. 3 would operate in a similar environment as weld No. 5 of the CRDM housing. Weld No. 4 would be exposed to a lower operating temperature than weld No. 5, however, TGSCC can still occur at 250 degrees Fahrenheit as evidenced by the Palisades previous operating experience with cracking identified in the seal housings that operate at even lower temperatures.

- Susceptible material – Welds No. 3 and No. 4 are composed of the same weld filler and base metal materials as weld No. 5 (e.g. weld filler material consistent with the type 316 stainless housing base metal). This material would be equally susceptible to TGSCC, as the type 347 stainless steel and weld filler materials used in the pre-2001 CRDM housing design that developed a through wall leak caused by TGSCC at weld No.3.

- Tensile stresses - While it is assumed that the corrective actions taken in response to the 2001 leak will reduce the potential for tensile stresses to exist on the inner surface of CRDM housings at welds No. 3 and No. 4, especially in light of repairs made to welds No. 3 and No. 4, it had not been conclusively demonstrated that these tensile stresses have been eliminated. As such, when evaluating welds No. 3 and No. 4 for applicability to the 2012 root cause, it was not reasonable to conclude that tensile stresses were not present, and therefore, the potential for TGSCC had been eliminated.

The 2012 RCR discussed manufacturing irregularities and misalignment between CRDM-24 and the support tube, seismic supports, and the associated reactor head penetration/CRDM nozzle as potential source of stresses leading to cracking. However, the RCR also stated that "based on the lack of cracking found in the other eight upper housings tested, the failed CRDM-24 upper housing contains an as-yet unidentified additional stress." Because the cause of the additional stress was not identified, the licensee had not established a basis in the RCR to exclude welds No. 3 and No. 4 from the extent of condition review (e.g. potential generic implications). In 2001, assumptions on crack growth rate and inspection intervals for welds No. 3 and No. 4 were made based on the information known at the time. The 2001 crack went through-wall after the CRDM was in service for 30 years and the cracking was widespread among the other CRDM housings. In 2012, the crack propagated through-wall after the CRDM was in service for 11 years and the cracking did not appear as widespread. Though TGSCC was a factor in both cracking events, there are still unknowns associated with the 2012 incident. The unknown additional stresses, as well as the time the CRDM was inservice before cracking in 2012, represent key differences as related to the cracking identified in 2001. In the 2012 RCR, the licensee did not consider these or other potential differences between the two incidents when determining not to include welds No. 3 and No. 4 in the evaluation and documentation of the generic implications of the root and contributing causes and therefore, did not provide a justification for excluding welds No. 3 and No.4 from this evaluation or corrective actions.

The inspectors identified that the licensee had not followed Procedure EN-LI-118 "Root Cause Evaluation," in the root cause review of the CRDM-24 leak as documented in report CR-PLP-2013-05623. Section 5.5 (12)e of EN-LI-118 required that the licensee "perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affect other SSC's." Additional details are provided in the procedure on how to conduct and document the evaluation. In this case, the inspectors identified that the licensee had not addressed or documented a basis in RCR CR-PLP-2013-05623 to exclude welds No. 3 and No. 4 from the generic factors discussed above that led to the 2012 leak in CRDM-24 (e.g. TGSCC at weld No. 5) to meet the procedural requirement. The licensee entered this issue into the corrective action program as CR-PLP-2013-01500. To restore compliance with the procedure, the licensee intended to revise the inspection plan to add additional corrective actions to inspect a sample of welds No. 3 and No. 4 for TGSCC during the upcoming refueling outage.

Analysis: The inspectors determined that the failure to adequately evaluate and document the generic implications of the cause of the cracking identified in CRDM-24 in accordance with the root cause procedure EN-LI-118 was a performance deficiency that warranted a significance evaluation. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. The inspectors also answered "yes" to the More-than-Minor screening question, "if left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, absent NRC identification, the licensee would not have completed further evaluations or inspections of CRDM housing welds, which could have resulted in additional CRDM housing failure and leakage by TGSCC. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609, Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a large component rupture.

The inspectors determined that the primary cause of the failure to adequately consider welds No. 3 and No. 4 in the generic implications section of the RCR related to the cross-cutting component of Human Performance, Decision Making, because licensee *staff did not use conservative assumptions in decision making.* Specifically, the licensee did not use conservative assumptions when excluding welds No. 3 and No. 4 as being susceptible to TGSCC and therefore include them in the generic implications section of the RCR. (Item H.1(b) of IMC 310).

Enforcement: During the inspection, the inspectors identified one NCV of NRC requirements:

Title 10 CFR Part 50, Appendix B, Criterion V "Instructions, Procedures and Drawings requires in part, activities affecting quality shall be prescribed by documented procedures and shall be accomplished in accordance with these procedures.

Procedure EN-LI-118 "Root Cause Evaluation Process," Revision 17 states:

- 5.5 (12)e: perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affects other SSCs , organizations or work processes. Use the two step process in accordance with attachment 9.7
- Attachment 9.7: Determine whether the occurrence/consequence (problem) is isolated, or whether it has broader (generic or common mode) implications. Achieve this by asking the following questions:
    - i. Could this happen to equipment that is similar in function, design, or service condition?
    - ii. Could this happen to a group of components? (components of the same construction or materials that could be similarly affected by one condition)
- Attachment 9.7: Document the results of the above considerations. Include the following items in the write up:
    - i. Generic Implications (Is this problem/ cause limited to this component/equipment, or does it apply to others as well)
    - ii. Existing broader (generic/common mode) considerations

- 5.5(15)(10)c&f: Document proposed corrective actions and due dates to address valid generic implications. If no corrective action is recommended for a valid generic implication then document the basis for this conclusion and any risk or consequence identified as a result of taking no action.

Contrary to the above, from February 24, 2013 through April 18, 2013, the licensee failed to accomplish activities affecting quality in accordance with procedure EN-LI-118, which was being implemented to correct a SCAQ. Specifically, the licensee failed to accomplish step 5.5 (12)e by not fully evaluating and documenting the existing broader (generic/common mode) considerations, extent of condition/cause associated with TGSCC at CRDM housing welds No. 3 and No. 4, including considering the susceptibility of the welds to TGSCC and performing subsequent inspections or evaluations.

The licensee intends to revise the inspection plan to add additional corrective actions to inspect a sample of welds No. 3 and No. 4 for TGSCC during the upcoming refueling outage.

Because of the very low safety significance and because the licensee entered this issue into their corrective action program (CR-PLP-2013-01500), it is being treated as a NCV consistent with Section 2.3.2 of the Enforcement Policy (NCV 05000255/2013003-xx Failure to Adequately Address the Generic Implications of the Cracking Identified in CRDM-24).

4OA5 Other Activities

.1 (Closed) Unresolved Item 05000255/2012012-02: Potential Inadequate Degradation Evaluation of CRDM Housings (This inspection is part of the additional inspections included in the Palisades Deviation letter)

During a Special Inspection performed in August 2012, NRC inspectors identified an issue, which could not be resolved without additional information (URI). This issue was associated with the rate of growth of the crack which created the through wall leak in CRDM-24, discovered on August 12, 2012. Identification of this crack growth rate is significant in determining appropriate intervals for future inspections to provide reasonable assurance that CRDM housing leakage will not recur.

Preliminary failure analysis data available at the time of the inspection indicated that the observed cracking was due to TGSCC. Cracking of this type is normally due to the presence of oxygen and chlorides at the location of the crack. When examining the fracture surface at the location the through-wall leak occurred, the licensee identified six concentric rings (beach marks) propagating in a radial direction from the ID out towards the OD of the housing. Beach marks are normally associated with fatigue failures and indicate the number of stress cycles from crack initiation to crack failure. In this case, there was no evidence that fatigue contributed to the failure. Despite the lack of evidence of fatigue, it was apparent that the crack, which resulted in the CRDM-24 leak, grew in increments. It was not, however, immediately apparent whether the increments were related to oxygen ingress (refueling outages) or temperature/pressure cycles (heatups/cooldowns).

At the time of the original inspection, five time intervals for through wall crack growth were under consideration. Two were based on literature crack growth data and three were based on interpretations of the beach marks. These time intervals were:

- Based on literature data, one contractor estimated that a 10% through wall flaw would require four years to reach 50% through wall.
- Based on literature data another contractor estimated the crack growth rate to be 2.1 x 10^{-5} in/hr or 0.18 in/yr. This is approximately three times faster than the crack growth rate proposed in the above mentioned rate.
- Based on the concept of oxygen ingress at refueling outages six cycles of 18 months duration would require nine years for the crack to grow through wall
- Based on the concept of temperature/pressure cycles, the plant experienced six cold shutdowns in approximately two years preceding the crack. This equates to two years for the crack to grow through wall.
- Based on the concept that oxygen is required for crack growth and that oxygen is rapidly purged from the CRDM housings due to leakage past the seals, crack growth occurs only during the first few weeks of operation following a refueling outage, followed by no growth for the remaining period of operation when oxygen concentrations are low. This equates to six oxygen ingress events (irrespective of time between events) for the crack to grow through wall.

NRC inspectors including technical experts from NRC Headquarters performed a follow-up inspection to determine if the assumptions made by the licensee were conservative and the planned actions bounded those conservative assumptions. The inspectors reviewed a variety of documents associated with crack growth and inspection intervals. The inspectors noted the following statements included in the RCR and vendor documents related to the determination of the appropriate crack growth rate:

- The laboratory conducting the failure analysis concluded, it could not be conclusively determined if the beach marks corresponded to refueling outages, (i.e., 18 month cycle) or shorter periods as occurred during outages over the past 24 months
- Palisades CRDM-21 leaked at weld No. 3 in 2001. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 failure. In calculating the crack growth rate of this crack, one contractor utilized an interval between beach marks, which is much shorter than refueling outages. The intervals used are consistent with plant thermal cycles in which oxygen may or may not have been admitted into the CRDMs.
- A CRDM housing at Ft Calhoun leaked at weld No. 5 in 1990. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 Palisades failure. In calculating the crack growth rate of this crack, Ft Calhoun stated that the beach marks were related to refueling cycles. Ft Calhoun also performed calculations indicating that the oxygen level at the location of the flaw did not change with time (including in response to refueling outages) because the CRDM housing was not vented. Ft Calhoun's evaluation indicated that oxygen levels at the vicinity of the crack would have begun to decline through diffusion and convection had the intervals between outages been much longer than 18 months. This is interpreted to mean that the beach marks at Ft Calhoun are in response to pressure/thermal cycles.
- In at least one instance, Palisades needed to repair the seals on a reactor coolant pump at a time other than an outage. This necessitated draining some of the water

from the reactor coolant system and venting (admitting oxygen into) the CRDM housing. This represented an additional oxygen ingress event not included when determination of time to cracking is based on refueling outages.

- In its inspection plan, Palisades stated that it will inspect all CRDM housings over the next four refueling outages, i.e., the interval between inspections is one refueling outage

Based on the above review, the inspectors noted that there were certain non conservative statements contained in the RCR and the inspection plan. These included:

- The crack growth rate based on refueling outages was understated. If oxygen ingress is related to beach marks, given the oxygen ingress event which occurred to repair reactor coolant pump seals, six beach marks would occur in a maximum of five refueling intervals rather than the six refueling intervals that were used to calculate the crack growth rate in the RCR.
- The crack growth rate based on heat up and cool down cycles is overstated. The value in the root cause is based on 11 months. While six shutdowns did occur at the plant in 11 months several of these events did not result in pressure/temperature changes of the reactor coolant system. The appropriate timeframe is 24 months rather than 11.
- The inspection plan contains a non conservative statement: "However, once the crack has been initiated it propagates over four to five operating cycles prior to going through wall." While this statement does reflect one of the proposed theories for crack growth, sufficient evidence to demonstrate reasonable assurance that this theory is correct, and thereby overcome the non-conservatism of this statement, was not provided.

Despite the existence of the non conservatisms stated above, the inspectors concluded:

- Sufficient evidence to conclusively determine the rate of crack growth does not exist.
- Crack growth based on pressure/temperature cycles is the most conservative of the potential crack growth mechanisms. In the absence of reasonable assurance of the correctness of less conservative mechanisms, through wall crack growth in two years must be utilized for regulatory purposes.
- The licensee has not formally committed to any of the crack growth mechanisms discussed.
- The licensee's inspection program includes inspection of all of the CRDM housings over the next four refueling outages. Approximately 25% of the housings will be inspected during each outage. The inspection of 25% of the CRDM housings each interval is sufficient to indicate that, in the event no indications are found during a given inspection, that the probability that flaws exist in other housings is extremely low. As such, it may be considered that the inspection of approximately 25% of the CRDM housings every refueling outage bounds all the crack growth rate mechanisms considered.

Overall, some weaknesses did exist in the site's assessment, but none of these issues arose above the level of a minor performance deficiency for the evaluations completed. With the corrective actions in place to monitor the CRDMs, the inspectors considered this approach to inspection to be both acceptable and sufficient justification to close this URI.

4OA6 Management Meetings

.2 Interim Exit Meetings

An interim exit was conducted for:

- The results of the selected issue follow-up inspection, with Mr. T. Vitali, Site Vice President on April 18, 2013.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

B. Davis, Engineering Director
O. Gustafson, Licensing Manager
T. Foudy, Engineering Supervisor
B. Williams, Engineer
B. Dotson, Licensing

LIST OF ITEMS OPENED, CLOSED, DISCUSSED

Closed

05000255/2012012-01	URI	TS for PCS Pressure Boundary Leakage
05000255/2012012-02	URI	Potential Inadequate Degradation Evaluation of CRDM Housings
05000255/2012012-03	URI	Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality

Opened and Discussed

None.

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

4OA2 Identification and Resolution of Problems

4OA5 Other Activities

LIST OF ACRONYMS USED

Hills, David

From: Hills, David
Sent: Friday, May 03, 2013 2:26 PM
To: Orth, Steven
Subject: RE: Palisades Report

Steve,

One other possible approach to the enforcement in question. Would the below approach make you more comfortable with it? Ties it a bit more closely with a specific requirement. I have the most recent Quality Assurance Manual dated January 2013. Would just need to find the revision in affect at the time of the violation which I suspect isn't any different in the pertinent parts. Just a thought. Elba still has to satisfy you with respect to differentiating the first finding from the second finding, but that would be the case whether the second finding had an associated violation or not.

10 CFR 50, Appendix B, Criterion II requires that the quality assurance program be documented by written policies, procedures, or instructions and shall be carried out throughout plant life in accordance with these policies, procedures, or instructions.

The Entergy Quality Assurance Program Manual, Revision ??, dated ?????, Section A.6.b prescribes that the corrective action program for significant conditions adverse to quality shall require cause determination and a corrective action plan that precludes repetition.

Procedure EN-LI-118, "Root Cause Evaluation Process," prescribes the licensee's quality assurance program controls for cause determination, and in part, corrective actions.

Procedure EN-LI-118 states . . . . . .

Contrary to the above, ...

- Dave

From: Sanchez Santiago, Elba
Sent: Friday, May 03, 2013 2:14 PM
To: Orth, Steven
Cc: Hills, David
Subject: Palisades Report
Importance: High

Steve,

Attached is the latest draft of the report. I incorporated some of the comments you made and made some changes to address others. You will notice there is still some mention of the corrective actions taken in 2001 in the write-up for the Proposed Criterion V. The reason for this is to address the arguments presented by the licensee in the white paper they provided us. I also included a paragraph that more clearly states what the differences between 2001 and 2012 are.

With regards to some of your comments related to the Criterion XVI issue, specifically the comment on corrective action, the list provided in the write-up is to provide a comparison of the information available at the time and different actions taken as they relate to weld #5. It is not meant to represent the proposed corrective actions to prevent recurrence. I made some changes to more clearly state this in a separate paragraph.

C 3/6

Feel free to contact me with questions or comments. I can be reached today at the office (630-829-9715). If you are reviewing the report at a later time, feel free to call me to help resolve any questions or concerns you may have. I can be reached at 787-236-9005.

Thanks,

Elba M. Sanchez Santiago

Reactor Engineer
RIII/ DRS/ EB1
630-829-9715

Hills, David

From: Hills, David
Sent: Monday, May 06, 2013 7:35 AM
To: Sanchez Santiago, Elba
Cc: Holmberg, Mel
Subject: Palisades Report Comments
Attachments: Comments on Draft Palisades Report.PDF

Elba,

My comments are attached. Once you have them incorporated and hear back from Steve Orth, you should be in a position to provide the draft to Pat Louden so he can make a decision on whether we move forward as indicated.

- Dave

C59

UNITED STATES
NUCLEAR REGULATORY COMMISSION
LISLE, IL 60532-4352

May XX, 2012

MEMORANDUM TO: Thomas Taylor
Senior Resident Inspector
Palisades Nuclear Plant

FROM: David Hills, Chief
Engineering Branch 3
Division of Reactor Safety

SUBJECT: PALISADES NUCLEAR PLANT
DRS INPUT TO INTEGRATED REPORT 05000255/2013002

Enclosed is the report input for the Palisades Nuclear Plant, Inspection Report 05000255/2013002. This report input documents completion of our review of Unresolved Items 05000255/2012012-01, "TS for PCS Pressure Boundary Leakage," 05000255/2012012-02, "Potential Inadequate Degradation Evaluation of CRDM Housings," and 05000255/2012012-03, "Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality." This report also completes one sample of the Problem Identification and Resolution, Selected Issue Follow-up inspection in accordance with IP 71152. I have reviewed this input to confirm compliance with Inspection Manual Chapter (IMC) 0612 and IMC 0305. This input is ready for inclusion into the integrated report and dissemination to the public.

Please input the following post Inspection Data into RPS:

Inspection Procedure	**Procedure Status** – see below: Incomplete, Complete, Complete by reference, Complete-full sample not available, Complete – opportunity to apply procedure not available, Not Applicable.	**Sample Size** – As documented in Scope Section If less than full sample size documented in the report input, the inspector must provide a justification below to enter into RPS and support the procedure status selected
71152	Complete	1

Inspection Report Item and Type (AV, FIN, NCV, URI or VIO)	**Cornerstone** (IE, MS, BI, EP, OR, PR, MISC)	**Cross Cutting Aspect** (H.n(i), P.n(i), S.n(i))	**Responsible Person/Owner**	**Procedure or TI** (71111.07T)	**RPS Branch Code (e.g. closeout responsibility)** EB1 3820 EB2 3870 EB3 3840 PST (RP) 3860 PSB (Safeguards) 3850 OB 3810
NCV-XXX	IE	n/a	E. Sanchez Santiago	71152	3820
NCV-XXX	IE	H.1(b)	E. Sanchez Santiago	71152	3820

Enclosure: Input to Inspection Report 05000255/2013002

cc w/encl: J. Giessner, Chief
C. Hernandez, Site Admin Assistant

CONTACT: E. Sanchez Santiago, DRS
(630) 829-9715

DOCUMENT NAME: G:\DRSIII\DRS\Work in Progress\-Palisades Input to DRP Report 2013 002 URI EMS.docx
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OFFICE	RIII		NRR		RIII		NRR	
NAME	ESanchezSantiago		DAlley		DHills		TLupold	
DATE	5/ /13							

OFFICIAL RECORD COPY

Cover Letter

X Green findings involving a violation were identified. Include the following:

Based on the results of this inspection, two NRC-identified findings of very low safety significance (Green) were identified. These findings were determined to involve a violation of NRC requirements. However, because of the very low safety significance and because the issues were entered into your corrective action program, the NRC is treating the issue as Non-Cited Violation, in accordance with Section 2.3.2 of the NRC Enforcement Policy.

TITLE PAGE

Inspectors: D. Alley, Senior Materials Engineer
E. Sanchez Santiago, Reactor Inspector

SUMMARY OF FINDINGS

A. NRC-Identified and Self-Revealed Findings

Cornerstones: Initiating Events

- Green. A self-revealing Green Finding with associated Non-Cited Violations (NCV) of 10 CFR Part 50, Appendix B, Criterion XVI and Technical Specification (TS) 3.4.13 Primary Coolant System (PCS) Operational Leakage, was identified for failure to take corrective actions to prevent recurrence of Control Rod Drive Mechanism (CRDM) cracking and leakage, a significant condition adverse to quality (SCAQ), and resulting in operation of the reactor with PCS pressure boundary leakage. Specifically, for Criterion XVI the licensee failed to include the internal CRDM housing weld build-up area within the scope of corrective actions taken for a 2001 CRDM through wall leak on CRDM-21, caused by transgranular stress corrosion cracking (TGSCC). Subsequently, a through wall leak recurred in the weld build-up area on CRDM-24 in 2012 due to TGSCC. As a result, the licensee operated with PCS pressure boundary leakage, which is not allowed by TS 3.4.13. Further, because the licensee was not aware that the leakage was PCS pressure boundary leakage, the licensee did not implement the associated TS action statement. The licensee replaced CRDM-24 upper housing and wrote CR-PLP-2013-01134. Additional corrective actions are described in NRC Inspection Report 05000255/2012012.

  The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone objective to limit the likelihood of events that upset plant stability. The issue was associated with the attribute of equipment performance. Specifically, the licensee did not take adequate corrective actions to prevent recurrence of leakage in CRDM housings, which represents pressure boundary leakage. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System Loss of Coolant Accident (LOCA) initiator contributor. The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the slow rate of change for leakage for this cracking mechanism and this type of material. Type 316 stainless steel material under TGSCC will experience leakage rates well below a small break LOCA, which would be observed through the crack, alerting operators to take action to shut down the plant prior to experiencing a component rupture. The cause of this finding, non-conservative decision making, occurred over ten years ago and is well outside of the nominal three year period in IMC 0612; and would not be indicative of current performance, unless there were other opportunities to identify the issue; therefore, the inspectors concluded this was not indicative of current performance. However more recently, the licensee exhibited non-conservative decision making with respect to addressing the potential for CRDM housing cracking and leakage during the recent root cause (Section 4OA2.3 (b.2) of this report), resulting in another finding. This cross-cutting aspect will be captured through the other finding. (Section 4OA2.3(b.1))

- Green. The inspectors identified a Finding with an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, for the licensee's failure to accomplish quality activities in accordance with the prescribed procedures. Specifically, the licensee failed to adequately evaluate and document the generic implications of the cause of the 2012 cracking identified in CRDM-24 in accordance with ~~root cause~~ procedure EN-LI-118, "Root Cause Evaluation." This issue was entered into the licensee's corrective action program under CR-PLP-2013-01500. Subsequently, the licensee decided to reinspection plus to add additional corrective actions to inspect a sample of welds No. 3 and 4 for TGSCC during the upcoming refueling outage.

The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because the inspectors answered "yes" to the More-than-Minor screening question, "if left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, absent NRC identification, the licensee would not have completed further evaluations or inspections of CRDM housing welds, which could have resulted in additional CRDM housing failure and leakage by TGSCC. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System LOCA initiator contributor. The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609, Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the slow rate of change for leakage for this cracking mechanism and this type of material. Type 316 stainless steel material under TGSCC will experience leakage rates well below a small break LOCA, which would be observed through the crack, alerting operators to take action to shut down the plant prior to experiencing a component rupture. The inspectors determined that the primary cause of the failure to adequately consider welds No. 3 and No. 4 in the generic implications section of the root cause report (RCR) related to the cross-cutting component of Human Performance, Decision Making, because licensee staff did not use conservative assumptions in decision making. Specifically, the licensee did not use conservative assumptions when excluding welds No. 3 and No. 4 as being susceptible to TGSCC when there was not enough information to exclude them from consideration. (Item H.1(b)). (Section 4OA2.3(b.2))

B. Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAILS

4. REACTOR SAFETY

4OA2 Identification and Resolution of Problems (71152)

.3 Selected Issue Follow-up Inspection: Through Wall Leakage of CRDM-24 (This inspection is part of the additional inspections ~~included~~ referenced in the Palisades Deviation letter)

a. Inspection Scope

On August 12, 2012, the licensee shut down the plant to investigate an increase in unidentified leakage. The source of the leakage was determined to be a crack in CRDM-24. The NRC dispatched a special inspection team (SIT) to review the CRDM-24 leakage event. The results of that inspection are provided in Inspection Report 05000255/2012012. The licensee completed an evaluation to determine the cause of the cracking (CR-PLP-2012-05623).

From March 4, 2013 to March 15, 2013, the inspectors completed one inspection sample regarding problem identification and resolution based upon review of the licensee's RCR contained in corrective action document CR-PLP-2012-05623. In addition, the inspectors performed reviews related to three Unresolved Items (URI) identified during the SIT inspection:

- URI 05000255/2012012-01 TS for PCS Pressure Boundary Leakage. (The closure of this URI is documented in section 4OA2.3 (b.1) of this report.)
- URI 05000255/2012012-02 Potential Inadequate Degradation Evaluation of CRDM Housings (The closure of this URI is documented in section 4OA5.1 of this report.)
- URI 05000255/2012012-03 Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality (The closure of this URI is documented in section 4OA2.3 (b.1) of this report.)

The inspectors reviewed the licensee's actions in accordance with performance attributes identified in IP 71152. Specifically, the inspectors reviewed licensee corrective action records to determine if: (1) the problems were accurately identified; (2) operability and reportability were adequately ascertained; (3) extent of condition and generic implications were appropriately addressed; (4) classification and prioritization of the problem were commensurate with safety significance; (5) root and contributing causes were identified; (6) corrective actions were appropriately focused to correct the problem; and (7) timely corrective actions were completed or proposed commensurate with the safety significance of the issues.

b. Findings

.1 Failure to Take Corrective Actions to Prevent Recurrence of CRDM Housing Cracking and Leakage

Introduction: A self-revealing Green Finding with associated NCV of 10 CFR Part 50, Appendix B, Criterion XVI and TS 3.4.13 PCS Operational Leakage, was identified for failure to take corrective actions to prevent recurrence of CRDM cracking and leakage, a SCAQ, and resulting in operation of the reactor with PCS pressure boundary leakage. Specifically, for Criterion XVI the licensee failed to include the internal CRDM housing weld build-up area within the scope of corrective actions taken for a 2001 CRDM through wall leak on CRDM-21 caused by TGSCC. Subsequently, a through wall leak recurred in the weld build-up area on CRDM-24 in 2012 due to TGSCC. As a result, the licensee operated with PCS pressure boundary leakage, which is not allowed by TS 3.4.13. Further, because the licensee was not aware that the leakage was PCS pressure boundary leakage, the licensee did not implement the associated TS action statement.

Description: In 2001, the licensee discovered a steam leak in the housing of CRDM-21 caused by a through-wall TGSCC at CRDM housing weld No. 3, which was located just below the weld build-up region (weld No. 5). Weld No. 5 consists of a weld material deposit applied to the inside diameter (ID) of the CRDM housing which provides for alignment of the CRDM. This issue was categorized as a SCAQ by the licensee (CPAL0102186) because it represented a break in the reactor system pressure boundary. The licensee's root cause evaluation was documented in RCR/C-PAL-01-02186 and concluded that the cracks in CRDM-21 were caused by TGSCC, which occurred in areas of heavy grinding or machining tool marks. Specifically, this leak was the result of an ID initiated, axially oriented, transgranular crack in the austenitic stainless steel housing material. The failure analysis performed in response to this event identified both axial and circumferential cracks associated with weld No. 3. Extent of condition inspections revealed additional, non-through wall cracks associated with weld No. 3 in 41 of the 44 remaining housings for a total of 42 of 45 housings containing cracks.

In response to the 2001 cracking, Palisades replaced all 45 CRDM housings with housings thought to be more resistant to cracking. Principle changes included:

- Elimination of weld No. 2,
- Relocation of weld No. 3 to a higher location thereby minimizing the deposition of crud in the gap between the weld and the bottom plate of the rack and pinion assembly,
- Reduction in residual stresses and cold work on welds by requiring better surface finishes, and
- Use of heat sink welding to reduce ID residual tensile stresses.

In January of 2002, an NRC SIT (reference IR 50-2555/01-15) reviewed the licensee proposed corrective actions associated with the through-wall leakage of the CRDM-21 housing caused by TGSCC. The 2001 RCR reviewed by the NRC stated the action to prevent recurrence was to "develop and implement an inspection plan to address areas and components identified in Attachment C-Extent of Condition. One of the components included in Attachment C was the CRDM. The recommended action was to perform volumetric inspection of the welds contained in the CRDM. Subsequently, the licensee decided to ~~changed the corrective actions and~~ exclude~~d~~ weld No. 5.

Was this really a change to the C.As? Or was it a more detailed implementation decision? If an overt change, then leave as is. Otherwise, make [illegible] change

Following the subsequent 2012 CRDM-24 leak, the licensee determined the leak occurred because of a through-wall crack adjacent to weld No. 5. The licensee formed a root cause team (RCT) staffed with licensee personnel and augmented with input from vendors. The root cause investigation was conducted in accordance with site procedure EN-LI-118 "Root Cause Evaluation Process" and was documented in root cause analysis report CR-PLP-2012-05623. In this report, the licensee's RCT determined that the probable cause of the cracking was:

> "Stresses in the weld build up area due to manufacturing irregularities and misalignments between CRDM-24 upper housing, support tube, and the associated reactor head penetration/CRDM nozzle. Based on lack of cracking found in the other eight upper housings tested, the failed CRDM-24 upper housing contains an as-yet unidentified additional stress."

The RCT also identified the following contributing cause:

> "TGSCC initiating within the internal weld build-up material of CRDM-24. The through wall crack initiated in the weld material and then propagated through the base metal until a leak developed in the outer diameter (OD) witness band region at the base of the ID weld build up.

This conclusion was based upon destructive and non destructive examinations (NDE) completed on a section of the failed housing, which included the through-wall flaw. The RCT also relied upon vendor technical reports assessing the results of the NDE as well as vendor calculations related to the stresses in the CRDM housings.

To determine the extent of condition, the licensee performed ultrasonic (UT) examinations of weld No. 5 on eight additional CRDM housings. The licensee selected these housings based on being in a similar location on the head as CRDM-24, and previous cracking having been identified in some of these housings prior to the replacement of the CRDM upper housings and seal housings in 2002. The inspectors concluded that this was an adequate sample for an initial extent of condition review based upon the concept that, in light of eight negative exams, the statistical probability of a flaw in the remaining CRDM housings was very low. Additionally, the licensee planned to conduct examinations of more housings during the next refueling outage.

The inspectors concluded that the licensee actions following the 2001 leak were not adequate because the appropriate actions to preclude recurrence were within the licensee's ability to foresee and implement. Specifically, the inspectors concluded that the licensee did not effectively implement corrective actions for the 2001 CRDM housing leak resulting in the 2012 CRDM-24 housing leak.

Licensee corrective actions taken in response to the 2001 event were limited to butt welds. The inspectors reviewed the licensee actions to determine if they had been sufficient to eliminate one of the three necessary factors to cause TGSCC on the CRDM housings: (1) a susceptible material, (2) a corrosive environment and (3) tensile stress. The inspectors identified that the licensee had failed to eliminate one or more of the necessary factors at weld No. 5 (which was not a butt weld) to preclude TGSCC in the replacement housing. Specifically:

- The licensee's 2001 RCR documented that weld No. 5 is exposed to essentially the same environment as the weld that experienced the cracking (corrosive environment remained unchanged).

- No analysis was completed on the stress conditions for weld No. 5 prior to approving the modified replacement housing design (the potential for residual tensile weld stresses on ID of CRDM surface was not ruled out by analysis and therefore, should have been considered).

- Fabrication restrictions to prohibit grinding were not applied to weld No. 5 (grinding promotes residual tensile stress state on ID of CRDM surface).

- Machining was performed on weld No. 5 during the fabrication process in order to achieve the dimensions and geometry specified in the design. This process induced cold work stresses in the weld.

- Material was changed from type 347 to type 316 stainless steel (both materials are essentially equally susceptible to TGSCC).

Also, in 1991, the Fort Calhoun plant had experienced through-wall leakage due to TGSCC at weld No. 5 of its CRDM housings (same housing design) and this operational experience had been reviewed by the licensee and dismissed. In the licensee's 2001 root cause evaluation, the licensee reviewed the weld build-up region failure by TGSCC at Fort Calhoun and concluded it would not occur at Palisades. This conclusion was based on the assumption that a higher oxygen environment (more aggressive environment) would exist in the Fort Calhoun housings than in the inservice ~~Palisades~~ its housings. However the licensee did not confirm this assumption, ~~nor did~~ the licensee perform additional testing to determine if the environment of ~~their~~ inservice housings was sufficiently benign to prevent TGSCC. The licensee's 2012 RCT ~~reached a similar conclusion and~~ Similarly documented that due to organizational/ programmatic weakness at Palisades, the 1991 Fort Calhoun operating ~~experience was not adequately~~ utilized to include inspection of the weld No. 5. The inspectors identified that the licensee had missed a key opportunity to implement effective corrective actions that could have prevented recurrence of the 2001 leakage event and elected not to pursue that aspect further. Specifically, in EA-EAR-2001-0426-01 the licensee considered fabricating the replacement housings with Inconel 600 material because it was much more resistant to TGSCC, but ultimately decided not to do so. Additionally, various vendor reports were generated related to this issue in the mid 2000's. Those reports documented the potential susceptibility of weld No. 5 to TGSCC based upon a review of the CRDM housing conditions and available operating experience. The reports also noted that weld No. 5 was not inspected in any of the housings in 2001. One report in 2003 noted that weld No. 5 should have been examined as part of the action from the 2001 events since it was similar to Fort Calhoun. The issuance of these documents represented another opportunity for the licensee to identify the susceptibility of weld No. 5 to TGSCC prior to the cracking in CRDM-24.

Similar to whose conclusion? sounds like 2012 RCT was agreeing with 2001 licensee assumption.

The inspectors concluded the Technical Specification corrective actions taken in response to the 2001 CRDM through wall leak from TGSCC, a SCAQ, were not effective to preclude repetition. In particular, a through wall leak did recur on a CRDM from TGSCC. This issue was within the licensee's ability to foresee and correct; therefore, the issue was a performance deficiency. During the 2012 NRC special inspection, the NRC identified an URI for the TS pressure boundary leak. LCO 3.4.13 does not allow any RCS pressure boundary leakage.

of the primary coolant pressure boundary.

In particular, TS Basis B3.4.13 "RCS Operational Leakage," explains that "No pressure boundary leakage from within the primary coolant pressure boundary is allowed, being indicative of material degradation. Leakage of this type is unacceptable as the leak itself could cause further deterioration, resulting in increased leakage. ...

Further, Action B, associated with this LCO, requires shutdown to mode 3 in ~~six~~ 6 hours and mode 5 in 36 hours for such leakage. The licensee determined the CRDM-24 leakage commenced on or around July 14, 2012, and the plant continued to operate in this condition until August 12, 2012. Because the licensee was not aware of the existence of pressure boundary leakage, it failed to shut down the unit in six hours for a pressure boundary leak as required by TS 3.4.13 Action B. The NRC previously assessed the site's action for increasing unidentified leakage as part of the SIT. The NRC determined, at the time of higher unidentified leakage, the site took appropriate actions to attempt to locate the leak, eventually shutting down around .3 gallons per minute (gpm) leakage (earlier than the TS value of 1 gpm value for unidentified leakage). and was
The licensee did not identify the source of the leakage as pressure boundary leakage until the shutdown on August 12, 2012, when a tour near the vessel head revealed the leaking housing. The pressure boundary leakage resulted in a TS violation due to the performance deficiency associated with the above mentioned Criterion XVI violation.

Based on the review discussed above, URIs 05000255/2012012-01 "TS for PCS Pressure Boundary Leakage" and 05000255/2012012-03 "Potential Failure to Take Corrective Actions to Prevent Recurrence of a Significant Condition Adverse to Quality" are closed.

Analysis: The inspectors determined that the licensee's failure to prevent recurrence of TGSCC of the CRDM housings (a SCAQ) that resulted in a violation of TS was a performance deficiency that warranted a significance evaluation. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone objective to limit the likelihood of events that upset plant stability. The issue was associated with the attribute of equipment performance. Specifically, the licensee did not take adequate corrective actions to prevent recurrence of leakage in CRDM housings, which represents pressure boundary leakage. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the slow rate of change for leakage for this cracking mechanism and this type of material. Type 316 stainless steel material under TGSCC will experience leakage rates well below a small break LOCA, which would be observed through the crack, alerting operators to take action to shut down the plant prior to experiencing a component rupture.

The cause of this finding, non-conservative decision making, occurred over ten years ago and is well outside of the nominal ~~3~~ three year period ~~in IMC 0612~~; and would not be indicative of current performance, unless there were other opportunities ~~to identify~~ the

We stated earlier that there were other opportunities. Were the other opportunities greater than 3 years ago?

issue; therefore, the inspectors concluded this was not indicative of current performance. However more recently, the licensee exhibited non-conservative decision making with respect to addressing the potential for CRDM housing cracking and leakage during the recent root cause (Section 4OA2.3 (b.2) of this report), resulting in another finding. This cross-cutting aspect will be captured through the other finding.

Enforcement: During this inspection, the inspectors identified two NCVs of NRC requirements:

Title 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that, for significant conditions adverse to quality, the cause of the condition is determined and corrective action taken to preclude repetition.

TS LCO 3.4.13 requires PCS operational leakage be limited to "No pressure boundary LEAKAGE" when in Modes 1 through 4.

Contrary to the above, as of August 12, 2012, the licensee had failed to take corrective actions to preclude repetition for a SCAQ. Specifically, on June 21, 2001, the licensee discovered a through wall leak in CRDM-21 due to TGSCC and failed to reasonably include weld No. 5 in the corrective actions which resulted in a subsequent through wall leak in CRDM-24 due to TGSCC.

Contrary to the above, on or around July 14, 2012, PCS pressure boundary leakage at CRDM-24 existed while in Mode 1. Further, because the licensee was not aware that the leakage was PCS pressure boundary leakage, the licensee did not implement the associated TS action statement.

As a result of the second through wall leak, the licensee took corrective actions, which included the development of an inspection plan that would inspect weld No. 5 every outage until all CRDM housings were inspected.

Because these violations were of very low safety significance and were entered into the licensee's corrective action program as CR-PLP-2013-01134, these violations are being treated as an NCVs, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000255/2013002-*xx*; Failure to Take Corrective Action to Prevent Recurrence of CRDM Pressure Boundary Leakage).

.2 Failure to Adequately Address the Generic Implications of the Cracking identified in CRDM-24

Introduction: The inspectors identified a Finding with an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, for the licensee's failure to accomplish quality activities in accordance with the prescribed procedures. Specifically, the licensee failed to adequately evaluate and document the generic implications of the cause of the cracking identified in CRDM-24 in accordance with ~~root cause~~ Procedure EN-LI-118, "Root Cause Evaluation." This issue was entered into the licensee's corrective action program under CR-PLP-2013-05623.

Description: As a result of the cracking identified in CRDM-24, which was characterized as a SCAQ, the licensee performed a root cause evaluation in accordance with Procedure EN-LI-118~~, "Root Cause Evaluation."~~ This procedure was identified as quality related and served to implement control pursuant to the licensee's quality assurance program. While reviewing the 2012 RCR (CR-PLP-2013-05623) related to the cracking identified in CRDM-24, the inspectors identified that the licensee had not appropriately considered the generic implications of the cracking in the extent of condition review. The licensee's proposed corrective actions, as a result of the 2012 RCR, narrowly focused on weld No. 5, instead of also including broader actions to ensure other CRDM housing welds were fit for their intended service life. These corrective actions consist of performing inspections of welds No. 5 on all CRDM housing.

On March 13, 2013, the inspectors requested that the licensee provide the bases for excluding other CRDM housing welds (weld No. 3 below weld No. 5 and weld No. 4 above weld No. 5) from the 2012 RCR scope of planned corrective actions. On March 29, 2013, the licensee provided additional information to justify excluding these welds from the scope of the corrective actions. The licensee credited the corrective actions associated with the modifications to the CRDM housing design completed in 2001 as the basis to exclude housing welds No. 3 and No. 4 from additional actions to identify the extent of TGSCC. The corrective actions taken in 2001 included performing heat sink welding, which is a methodology used to reduce the stresses on the inner ID of the weld. The licensee also changed the design to reduce design stresses at weld No. 3 and specified a smoother surface finish (RMS 125) to reduce potential crack initiation points. The licensee stated that these actions would produce compressive stresses on the ID of welds No. 3 and No. 4 making them immune from cracking. The inspectors acknowledged that these actions would reduce the tensile stress at the ID surface and thus reduce the probability of initiating TGSCC. However, the information provided did not demonstrate that TGSCC would not occur because it did not demonstrate that tensile stress would be eliminated at the ID surface during operation.

The inspectors identified that the three factors required for TGSCC could still be present at welds No. 3 and No. 4 as follows:

- Corrosive environment – Weld No. 3 would operate in a similar environment as weld No. 5 of the CRDM housing. Weld No. 4 would be exposed to a lower operating temperature than weld No. 5, however, TGSCC can still occur at 250 degrees Fahrenheit as evidenced by the Palisades previous operating experience with cracking identified in the seal housings that operate at even lower temperatures.

- Susceptible material – Welds No. 3 and No. 4 are composed of the same weld filler and base metal materials as weld No. 5 (e.g. weld filler material consistent with the type 316 stainless housing base metal). This material would be equally susceptible to TGSCC, as the type 347 stainless steel and weld filler materials used in the pre-2001 CRDM housing design that developed a through wall leak caused by TGSCC at weld No.3.

- Tensile stresses - While it is assumed that the corrective actions taken in response to the 2001 leak will reduce the potential for tensile stresses to exist on the inner surface of CRDM housings at welds No. 3 and No. 4, especially in light of repairs made to welds No. 3 and No. 4, it had not been conclusively demonstrated that these tensile stresses have been eliminated. As such, when evaluating welds No. 3 and No. 4 for applicability to the 2012 root cause, it was not reasonable to conclude that tensile stresses were not present, and therefore, the potential for TGSCC had been eliminated.

The 2012 RCR discussed manufacturing irregularities and misalignment between CRDM-24 and the support tube, seismic supports, and the associated reactor head penetration/CRDM nozzle as potential source of stresses leading to cracking. However, the RCR also stated that "based on the lack of cracking found in the other eight upper housings tested, the failed CRDM-24 upper housing contains an as-yet unidentified additional stress." Because the cause of the additional stress was not identified, the licensee had not established a basis in the RCR to exclude welds No. 3 and No. 4 from the extent of condition review (e.g. potential generic implications). In 2001, assumptions on crack growth rate and inspection intervals for welds No. 3 and No. 4 were made based on the information known at the time. The 2001 crack went through-wall after the CRDM was in service for 30 years and the cracking was widespread among the other CRDM housings. In 2012, the crack propagated through-wall after the CRDM was in service for 11 years and the cracking did not appear as widespread. Though TGSCC was a factor in both cracking events, there are still unknowns associated with the 2012 incident. The unknown additional stresses, as well as the time the CRDM was inservice before cracking in 2012, represent key differences as related to the cracking identified in 2001. In the 2012 RCR, the licensee did not consider these or other potential differences between the two incidents when determining not to include welds No. 3 and No. 4 in the evaluation and documentation of the generic implications of the root and contributing causes and therefore, did not provide a justification for excluding welds No. 3 and No.4 from this evaluation or corrective actions.

The inspectors identified that the licensee had not followed Procedure EN-LI-118 ~~"Root Cause Evaluation,"~~ in the root cause review of the CRDM-24 leak as documented in report CR-PLP-2013-05623. Section 5.5 (12)e of EN-LI-118 required that the licensee "perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affect other SSC~~'~~s." Additional details are provided in the procedure on how to conduct and document the evaluation. In this case, the inspectors identified that the licensee had not addressed or documented a basis in RCR CR-PLP-2013-05623 to exclude welds No. 3 and No. 4 from the generic factors discussed above that led to the 2012 leak in CRDM-24 (e.g. TGSCC at weld No. 5) ~~to meet the procedural requirement~~. The licensee entered this issue into the corrective action program as CR-PLP-2013-01500. ~~To restore compliance with the procedure~~, the licensee ~~intended~~ decided to revise the inspection plan to add additional corrective actions to inspect a sample of welds No. 3 and No. 4 for TGSCC during the upcoming subsequently refueling outage.

Analysis: The inspectors determined that the failure to adequately evaluate and document the generic implications of the cause of the cracking identified in CRDM-24 in accordance with the root cause procedure EN-LI-118 was a performance deficiency that warranted a significance evaluation. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. The inspectors also answered "yes" to the More-than-Minor screening question, "if left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, absent NRC identification, the licensee would not have completed further evaluations or inspections of CRDM housing welds, which could have resulted in additional CRDM housing failure and leakage by TGSCC. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609, Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a large component rupture.

The inspectors determined that the primary cause of the failure to adequately consider welds No. 3 and No. 4 in the generic implications section of the RCR related to the cross-cutting component of Human Performance, Decision Making, because licensee staff did not use conservative assumptions in decision making. Specifically, the licensee did not use conservative assumptions when excluding welds No. 3 and No. 4 as being susceptible to TGSCC and therefore include them in the generic implications section of the RCR. (Item H.1(b) of IMC 310).

Enforcement: During the inspection, the inspectors identified one NCV of NRC requirements:

Title 10 CFR Part 50, Appendix B, Criterion V "Instructions, Procedures and Drawings requires in part, activities affecting quality shall be prescribed by documented procedures and shall be accomplished in accordance with these procedures.

Procedure EN-LI-118 "Root Cause Evaluation Process," Revision 17 states:

- 5.5 (12)e: perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affects other SSCs , organizations or work processes. Use the two step process in accordance with attachment 9.7
- Attachment 9.7: Determine whether the occurrence/consequence (problem) is isolated, or whether it has broader (generic or common mode) implications. Achieve this by asking the following questions:
    - i. Could this happen to equipment that is similar in function, design, or service condition?
    - ii. Could this happen to a group of components? (components of the same construction or materials that could be similarly affected by one condition)
- Attachment 9.7: Document the results of the above considerations. Include the following items in the write up:
    - i. Generic Implications (Is this problem/ cause limited to this component/equipment, or does it apply to others as well)
    - ii. Existing broader (generic/common mode) considerations

- 5.5(15)(10)c&f: Document proposed corrective actions and due dates to address valid generic implications. If no corrective action is recommended for a valid generic implication then document the basis for this conclusion and any risk or consequence identified as a result of taking no action.

Contrary to the above, from February 24, 2013 through April 18, 2013, the licensee failed to accomplish activities affecting quality in accordance with procedure EN-LI-118, which was being implemented to correct a SCAQ. Specifically, the licensee failed to accomplish step 5.5 (12)e by not fully evaluating and documenting the existing broader (generic/common mode) considerations, extent of condition/cause associated with TGSCC at CRDM housing welds No. 3 and No. 4, including considering the susceptibility of the welds to TGSCC and the need to perform~~ing~~ subsequent inspections or evaluations.

Subsequently, the licensee ~~intends~~ decided to revise the inspection plan to add additional corrective actions to inspect a sample of welds No. 3 and No. 4 for TGSCC during the upcoming refueling outage.

Because of the very low safety significance and because the licensee entered this issue into their corrective action program (CR-PLP-2013-01500), it is being treated as a NCV consistent with Section 2.3.2 of the Enforcement Policy (NCV 05000255/2013003-xx Failure to Adequately Address the Generic Implications of the Cracking Identified in CRDM-24).

4OA5 Other Activities

.1 (Closed) Unresolved Item 05000255/2012012-02: Potential Inadequate Degradation Evaluation of CRDM Housings (This inspection is part of the additional inspections included in the Palisades Deviation letter)

During a Special Inspection performed in August 2012, NRC inspectors identified an issue, which could not be resolved without additional information (URI). This issue was associated with the rate of growth of the crack which created the through wall leak in CRDM-24, discovered on August 12, 2012. Identification of this crack growth rate is significant in determining appropriate intervals for future inspections to provide reasonable assurance that CRDM housing leakage will not recur.

Preliminary failure analysis data available at the time of the inspection indicated that the observed cracking was due to TGSCC. Cracking of this type is normally due to the presence of oxygen and chlorides at the location of the crack. When examining the fracture surface at the location the through-wall leak occurred, the licensee identified six concentric rings (beach marks) propagating in a radial direction from the ID out towards the OD of the housing. Beach marks are normally associated with fatigue failures and indicate the number of stress cycles from crack initiation to crack failure. In this case, there was no evidence that fatigue contributed to the failure. Despite the lack of evidence of fatigue, it was apparent that the crack, which resulted in the CRDM-24 leak, grew in increments. It was not, however, immediately apparent whether the increments were related to oxygen ingress (refueling outages) or temperature/pressure cycles (heatups/cooldowns).

At the time of the original inspection, five time intervals for through wall crack growth were under consideration. Two were based on literature crack growth data and three were based on interpretations of the beach marks. These time intervals were:

- Based on literature data, one contractor estimated that a 10% through wall flaw would require four years to reach 50% through wall.
- Based on literature data another contractor estimated the crack growth rate to be 2.1 x 10^{-5} in/hr or 0.18 in/yr. This is approximately three times faster than the crack growth rate proposed in the above mentioned rate.
- Based on the concept of oxygen ingress at refueling outages six cycles of 18 months duration would require nine years for the crack to grow through wall
- Based on the concept of temperature/pressure cycles, the plant experienced six cold shutdowns in approximately two years preceding the crack. This equates to two years for the crack to grow through wall.
- Based on the concept that oxygen is required for crack growth and that oxygen is rapidly purged from the CRDM housings due to leakage past the seals, crack growth occurs only during the first few weeks of operation following a refueling outage, followed by no growth for the remaining period of operation when oxygen concentrations are low. This equates to six oxygen ingress events (irrespective of time between events) for the crack to grow through wall.

NRC inspectors including technical experts from NRC Headquarters performed a follow-up inspection to determine if the assumptions made by the licensee were conservative and the planned actions bounded those conservative assumptions. The inspectors reviewed a variety of documents associated with crack growth and inspection intervals. The inspectors noted the following statements included in the RCR and vendor documents related to the determination of the appropriate crack growth rate:

- The laboratory conducting the failure analysis concluded, it could not be conclusively determined if the beach marks corresponded to refueling outages, (i.e., 18 month cycle) or shorter periods as occurred during outages over the past 24 months
- Palisades CRDM-21 leaked at weld No. 3 in 2001. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 failure. In calculating the crack growth rate of this crack, one contractor utilized an interval between beach marks, which is much shorter than refueling outages. The intervals used are consistent with plant thermal cycles in which oxygen may or may not have been admitted into the CRDMs.
- A CRDM housing at Ft Calhoun leaked at weld No. 5 in 1990. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 Palisades failure. In calculating the crack growth rate of this crack, Ft Calhoun stated that the beach marks were related to refueling cycles. Ft Calhoun also performed calculations indicating that the oxygen level at the location of the flaw did not change with time (including in response to refueling outages) because the CRDM housing was not vented. Ft Calhoun's evaluation indicated that oxygen levels at the vicinity of the crack would have begun to decline through diffusion and convection had the intervals between outages been much longer than 18 months. This is interpreted to mean that the beach marks at Ft Calhoun are in response to pressure/thermal cycles.
- In at least one instance, Palisades needed to repair the seals on a reactor coolant pump at a time other than an outage. This necessitated draining some of the water

from the reactor coolant system and venting (admitting oxygen into) the CRDM housing. This represented an additional oxygen ingress event not included when determination of time to cracking is based on refueling outages.

- In its inspection plan, Palisades stated that it will inspect all CRDM housings over the next four refueling outages, i.e., the interval between inspections is one refueling outage

Based on the above review, the inspectors noted that there were certain non conservative statements contained in the RCR and the inspection plan. These included:

- The crack growth rate based on refueling outages was understated. If oxygen ingress is related to beach marks, given the oxygen ingress event which occurred to repair reactor coolant pump seals, six beach marks would occur in a maximum of five refueling intervals rather than the six refueling intervals that were used to calculate the crack growth rate in the RCR.
- The crack growth rate based on heat up and cool down cycles is overstated. The value in the root cause is based on 11 months. While six shutdowns did occur at the plant in 11 months several of these events did not result in pressure/temperature changes of the reactor coolant system. The appropriate timeframe is 24 months rather than 11.
- The inspection plan contains a non conservative statement: "However, once the crack has been initiated it propagates over four to five operating cycles prior to going through wall." While this statement does reflect one of the proposed theories for crack growth, sufficient evidence to demonstrate reasonable assurance that this theory is correct, and thereby overcome the non-conservatism of this statement, was not provided.

Despite the existence of the non conservatisms stated above, the inspectors concluded:

- Sufficient evidence to conclusively determine the rate of crack growth does not exist.
- Crack growth based on pressure/temperature cycles is the most conservative of the potential crack growth mechanisms. In the absence of reasonable assurance of the correctness of less conservative mechanisms, through wall crack growth in two years must be utilized for regulatory purposes.
- The licensee has not formally committed to any of the crack growth mechanisms discussed.
- The licensee's inspection program includes inspection of all of the CRDM housings over the next four refueling outages. Approximately 25% of the housings will be inspected during each outage. The inspection of 25% of the CRDM housings each interval is sufficient to indicate that, in the event no indications are found during a given inspection, that the probability that flaws exist in other housings is extremely low. As such, it may be considered that the inspection of approximately 25% of the CRDM housings every refueling outage bounds all the crack growth rate mechanisms considered.

Overall, some weaknesses did exist in the site's assessment, but none of these issues arose above the level of a minor performance deficiency for the evaluations completed. With the corrective actions in place to monitor the CRDMs, the inspectors considered this approach to inspection to be both acceptable and sufficient justification to close this URI.

4OA6 Management Meetings

.2 Interim Exit Meetings

An interim exit was conducted for:

- The results of the selected issue follow-up inspection, with Mr. T. Vitali, Site Vice President on April 18, 2013.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

B. Davis, Engineering Director
O. Gustafson, Licensing Manager
T. Foudy, Engineering Supervisor
B. Williams, Engineer
B. Dotson, Licensing

LIST OF ITEMS OPENED, CLOSED, DISCUSSED

Closed

05000255/2012012-01	URI	TS for PCS Pressure Boundary Leakage
05000255/2012012-02	URI	Potential Inadequate Degradation Evaluation of CRDM Housings
05000255/2012012-03	URI	Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality

Opened and Discussed

None.

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

4OA2 Identification and Resolution of Problems

4OA5 Other Activities

LIST OF ACRONYMS USED

Hills, David

From: Holmberg, Mel
Sent: Monday, May 06, 2013 11:12 AM
To: Orth, Steven
Cc: Sanchez Santiago, Elba; Giessner, John; Hills, David
Subject: Palisades Report Input- URI Closure CRDM-24
Attachments: Palisades Input to DRP Report 2013 002 URI EMS 5-6msh.docx

Steve,

The attached is the most recent version of the Palisades report with the violations proposed associated with the CRDM-24 housing leaks. This version has all of the review comments including yours incorporated. If you have any further comments please let us know today if possible,

Thanks,

Mel

C40

UNITED STATES
NUCLEAR REGULATORY COMMISSION
LISLE, IL 60532-4352

May XX, 2012

MEMORANDUM TO: Thomas Taylor
Senior Resident Inspector
Palisades Nuclear Plant

FROM: David Hills, Chief
Engineering Branch 3
Division of Reactor Safety

SUBJECT: PALISADES NUCLEAR PLANT
DRS INPUT TO INTEGRATED REPORT 05000255/2013002

Enclosed is the report input for the Palisades Nuclear Plant, Inspection Report 05000255/2013002. This report input documents completion of our review of Unresolved Items 05000255/2012012-01, "TS for PCS Pressure Boundary Leakage," 05000255/2012012-02, "Potential Inadequate Degradation Evaluation of CRDM Housings," and 05000255/2012012-03, "Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality." This report also completes one sample of the Problem Identification and Resolution, Selected Issue Follow-up inspection in accordance with IP 71152. I have reviewed this input to confirm compliance with Inspection Manual Chapter (IMC) 0612 and IMC 0305. This input is ready for inclusion into the integrated report and dissemination to the public.

Please input the following post Inspection Data into RPS:

Inspection Procedure	**Procedure Status** – see below: Incomplete, Complete, Complete by reference, Complete-full sample not available, Complete – opportunity to apply procedure not available, Not Applicable.	**Sample Size** – As documented in Scope Section If less than full sample size documented in the report input, the inspector must provide a justification below to enter into RPS and support the procedure status selected
71152	Complete	1

Inspection Report Item and Type (AV, FIN, NCV, URI or VIO)	**Cornerstone** (IE, MS, BI, EP, OR, PR, MISC)	**Cross Cutting Aspect** (H.n(i), P.n(i), S.n(i))	**Responsible Person/Owner**	**Procedure or TI** (71111.07T)	**RPS Branch Code (e.g. closeout responsibility)** EB1 3820 EB2 3870 EB3 3840 PST (RP) 3860 PSB (Safeguards) 3850 OB 3810
NCV-XXX	IE	n/a	E. Sanchez Santiago	71152	3820
NCV-XXX	IE	H.1(b)	E. Sanchez Santiago	71152	3820

Enclosure: Input to Inspection Report 05000255/2013002

cc w/encl: J. Giessner, Chief
C. Hernandez, Site Admin Assistant

CONTACT: E. Sanchez Santiago, DRS
(630) 829-9715

DOCUMENT NAME: G:\DRSIII\DRS\Work in Progress\-Palisades Input to DRP Report 2013 002 URI EMS.docx
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OFFICE	RIII		NRR		RIII		NRR	
NAME	ESanchezSantiago		DAlley		DHills		TLupold	
DATE	5/ /13							

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Cover Letter

X Green findings involving a violation were identified. Include the following:

Based on the results of this inspection, two NRC-identified findings of very low safety significance (Green) were identified. These findings were determined to involve a violation of NRC requirements. However, because of the very low safety significance and because the issues were entered into your corrective action program, the NRC is treating the issue as Non-Cited Violation, in accordance with Section 2.3.2 of the NRC Enforcement Policy.

TITLE PAGE

Inspectors: D. Alley, Senior Materials Engineer
E. Sanchez Santiago, Reactor Inspector

SUMMARY OF FINDINGS

A. NRC-Identified and Self-Revealed Findings

Cornerstones: Initiating Events

- Green. A self-revealing Green Finding with associated Non-Cited Violations (NCV) of 10 CFR Part 50, Appendix B, Criterion XVI and Technical Specification (TS) 3.4.13 Primary Coolant System (PCS) Operational Leakage, was identified for failure to take corrective actions to prevent recurrence of Control Rod Drive Mechanism (CRDM) cracking and leakage, a significant condition adverse to quality (SCAQ), and resulting in operation of the reactor with PCS pressure boundary leakage. Specifically, for Criterion XVI the licensee failed to include the internal CRDM housing weld build-up area within the scope of corrective actions taken for a 2001 CRDM through wall leak on CRDM-21, caused by transgranular stress corrosion cracking (TGSCC). Subsequently, a through wall leak recurred in the weld build-up area on CRDM-24 in 2012 due to TGSCC. As a result, the licensee operated with PCS pressure boundary leakage, which is not allowed by TS 3.4.13. Further, because the licensee was not aware that the leakage was PCS pressure boundary leakage, the licensee did not implement the associated TS action statement. The licensee replaced CRDM-24 upper housing and wrote CR-PLP-2013-01134. Additional corrective actions are described in NRC Inspection Report 05000255/2012012.

  The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone objective to limit the likelihood of events that upset plant stability. The issue was associated with the attribute of equipment performance. Specifically, the licensee did not take adequate corrective actions to prevent recurrence of leakage in CRDM housings, which represents pressure boundary leakage. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System Loss of Coolant Accident (LOCA) initiator contributor. The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the slow rate of change for leakage for this cracking mechanism and this type of material. Type 316 stainless steel material under TGSCC will experience leakage rates well below a small break LOCA, which would be observed through the crack, alerting operators to take action to shut down the plant prior to experiencing a component rupture. The cause of this finding, non-conservative decision making, occurred over ten years ago and is well outside of the nominal three year period in IMC 0612; and would not be indicative of current performance, unless there were other opportunities to identify the issue; therefore, the inspectors concluded this was not indicative of current performance. However more recently, the licensee exhibited non-conservative decision making with respect to addressing the potential for CRDM housing cracking and leakage during the recent root cause (Section 4OA2.3 (b.2) of this report), resulting in another finding. This cross-cutting aspect will be captured through the other finding. (Section 4OA2.3(b.1))

- Green. The inspectors identified a Finding with an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, for the licensee's failure to accomplish quality activities in accordance with the prescribed procedures. Specifically, the licensee failed to adequately evaluate and document the generic implications of the cause of the 2012 cracking identified in CRDM-24 in accordance with Procedure EN-LI-118 "Root Cause Evaluation." This issue was entered into the licensee's corrective action program under CR-PLP-2013-01500. Subsequently, the licensee decided to revise the inspection plan to add additional corrective actions to inspect a sample of welds No. 2 and No. 4 for transgranular stess corrosion cracking during the upcoming refueling outage.

  The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because the inspectors answered "yes" to the More-than-Minor screening question, "if left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, absent NRC identification, the licensee would not have completed further evaluations or inspections of CRDM housing welds, which could have resulted in additional CRDM housing failure and leakage by TGSCC. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System LOCA initiator contributor. The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609, Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a

LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the slow rate of change for leakage for this cracking mechanism and this type of material. Type 316 stainless steel material under TGSCC will experience leakage rates well below a small break LOCA, which would be observed through the crack, alerting operators to take action to shut down the plant prior to experiencing a component rupture. The inspectors determined that the primary cause of the failure to adequately consider welds No. 3 and No. 4 in the generic implications section of the root cause report (RCR) related to the cross-cutting component of Human Performance, Decision Making, because licensee staff did not use conservative assumptions in decision making. Specifically, the licensee did not use conservative assumptions when excluding welds No. 3 and No. 4 as being susceptible to TGSCC when there was not enough information to exclude them from consideration. (Item H.1(b)). (Section 4OA2.3(b.2))

B. Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAILS

4. REACTOR SAFETY

4OA2 Identification and Resolution of Problems (71152)

.3 Selected Issue Follow-up Inspection: Through Wall Leakage of CRDM-24 (This inspection is part of the additional inspections referenced in the Palisades Deviation letter.)

a. Inspection Scope

On August 12, 2012, the licensee shut down the plant to investigate an increase in unidentified leakage. The source of the leakage was determined to be a crack in CRDM-24. The NRC dispatched a special inspection team (SIT) to review the CRDM-24 leakage event. The results of that inspection are provided in Inspection Report 05000255/2012012. The licensee completed an evaluation to determine the cause of the cracking (CR-PLP-2012-05623).

From March 4, 2013 to March 15, 2013, the inspectors completed one inspection sample regarding problem identification and resolution based upon review of the licensee's RCR contained in corrective action document CR-PLP-2012-05623. In addition, the inspectors performed reviews related to three Unresolved Items (URI) identified during the SIT inspection:

- URI 05000255/2012012-01 Technical Specification (TS) for PCS Pressure Boundary Leakage. (The closure of this URI is documented in section 4OA2.3 (b.1) of this report.)
- URI 05000255/2012012-02 Potential Inadequate Degradation Evaluation of CRDM Housings (The closure of this URI is documented in section 4OA5.1 of this report.)
- URI 05000255/2012012-03 Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality (The closure of this URI is documented in section 4OA2.3 (b.1) of this report.)

The inspectors reviewed the licensee's actions in accordance with performance attributes identified in IP 71152. Specifically, the inspectors reviewed licensee corrective action records to determine if: (1) the problems were accurately identified; (2) operability and reportability were adequately ascertained; (3) extent of condition and generic implications were appropriately addressed; (4) classification and prioritization of the problem were commensurate with safety significance; (5) root and contributing causes were identified; (6) corrective actions were appropriately focused to correct the problem; and (7) timely corrective actions were completed or proposed commensurate with the safety significance of the issues.

b. Findings

.1 Failure to Take Corrective Actions to Prevent Recurrence of CRDM Housing Cracking and Leakage

Introduction: A self-revealing Green Finding with associated NCV of 10 CFR Part 50, Appendix B, Criterion XVI and TS 3.4.13 PCS Operational Leakage, was identified for failure to take corrective actions to prevent recurrence of CRDM cracking and leakage, a SCAQ, and resulting in operation of the reactor with PCS pressure boundary leakage. Specifically, for Criterion XVI the licensee failed to include the internal CRDM housing weld build-up area within the scope of corrective actions taken for a 2001 CRDM through wall leak on CRDM-21 caused by TGSCC. Subsequently, a through wall leak recurred in the weld build-up area on CRDM-24 in 2012 due to TGSCC. As a result, the licensee operated with PCS pressure boundary leakage, which is not allowed by TS 3.4.13. Further, because the licensee was not aware that the leakage was PCS pressure boundary leakage, the licensee did not implement the associated TS action statement.

Description: In 2001, the licensee discovered a steam leak in the housing of CRDM-21 caused by a through-wall TGSCC at CRDM housing weld No. 3, which was located just below the weld build-up region (weld No. 5). Weld No. 5 consists of a weld material deposit applied to the inside diameter (ID) of the CRDM housing which provides for alignment of the CRDM. This issue was categorized as a SCAQ by the licensee (CPAL0102186) because it represented a break in the reactor system pressure boundary. The licensee's root cause evaluation was documented in RCR/C-PAL-01-02186 and concluded that the cracks in CRDM-21 were caused by TGSCC, which occurred in areas of heavy grinding or machining tool marks. Specifically, this leak was the result of an ID initiated, axially oriented, transgranular crack in the austenitic stainless steel housing material. The failure analysis performed in response to this event identified both axial and circumferential cracks associated with weld No. 3. Extent of condition inspections revealed additional, non-through wall cracks associated with weld No. 3 in 41 of the 44 remaining housings for a total of 42 of 45 housings containing cracks.

In response to the 2001 cracking, Palisades replaced all 45 CRDM housings with housings thought to be more resistant to cracking. Principle changes included:

- Elimination of weld No. 2,
- Relocation of weld No. 3 to a higher location thereby minimizing the deposition of crud in the gap between the weld and the bottom plate of the rack and pinion assembly,
- Reduction in residual stresses and cold work on welds by requiring better surface finishes, and
- Use of heat sink welding to reduce ID residual tensile stresses.

In January of 2002, an NRC SIT (reference IR 50-2555/01-15) reviewed the licensee proposed corrective actions associated with the through-wall leakage of the CRDM-21 housing caused by TGSCC. The 2001 RCR reviewed by the NRC stated the action to prevent recurrence was to "develop and implement an inspection plan to address areas and components identified in Attachment C-Extent of Condition. One of the components included in Attachment C was the CRDM. The recommended action was to perform volumetric inspection of the welds contained in the CRDM. Subsequently, the licensee decided to change exclude weld No. 5.

Following the subsequent 2012 CRDM-24 leak, the licensee determined the leak occurred because of a through-wall crack adjacent to weld No. 5. The licensee formed a root cause team (RCT) staffed with licensee personnel and augmented with input from vendors. The root cause investigation was conducted in accordance with site procedure EN-LI-118 "Root Cause Evaluation Process" and was documented in root cause analysis report CR-PLP-2012-05623. In this report, the licensee's RCT determined that the probable cause of the cracking was:

> "Stresses in the weld build up area due to manufacturing irregularities and misalignments between CRDM-24 upper housing, support tube, and the associated reactor head penetration/CRDM nozzle. Based on lack of cracking found in the other eight upper housings tested, the failed CRDM-24 upper housing contains an as-yet unidentified additional stress."

The RCT also identified the following contributing cause:

> "TGSCC initiating within the internal weld build-up material of CRDM-24. The through wall crack initiated in the weld material and then propagated through the base metal until a leak developed in the outer diameter (OD) witness band region at the base of the ID weld build up.

This conclusion was based upon destructive and non destructive examinations (NDE) completed on a section of the failed housing, which included the through-wall flaw. The RCT also relied upon vendor technical reports assessing the results of the NDE as well as vendor calculations related to the stresses in the CRDM housings.

To determine the extent of condition, the licensee performed ultrasonic (UT) examinations of weld No. 5 on eight additional CRDM housings. The licensee selected these housings based on being in a similar location on the head as CRDM-24, and previous cracking having been identified in some of these housings prior to the replacement of the CRDM upper housings and seal housings in 2002. The inspectors concluded that this was an adequate sample for an initial extent of condition review based upon the concept that, in light of eight negative exams, the statistical probability of a flaw in the remaining CRDM housings was very low. Additionally, the licensee planned to conduct examinations of more housings during the next refueling outage.

The inspectors concluded that the licensee actions following the 2001 leak were not adequate because the appropriate actions to preclude recurrence were within the licensee's ability to foresee and implement. Specifically, the inspectors concluded that the licensee did not effectively implement corrective actions for the 2001 CRDM housing leak resulting in the 2012 CRDM-24 housing leak.

Licensee corrective actions taken in response to the 2001 event were limited to butt welds. The inspectors reviewed the licensee actions to determine if they had been sufficient to eliminate one of the three necessary factors to cause TGSCC on the CRDM housings: (1) a susceptible material, (2) a corrosive environment and (3) tensile stress. The inspectors identified that the licensee had failed to eliminate one or more of the necessary factors at weld No. 5 (which was not a butt weld) to preclude TGSCC in the replacement housing. Specifically:

- The licensee's 2001 RCR documented that weld No. 5 is exposed to essentially the same environment as the weld that experienced the cracking (corrosive environment remained unchanged).

- No analysis was completed on the stress conditions for weld No. 5 prior to approving the modified replacement housing design (the potential for residual tensile weld stresses on ID of CRDM surface was not ruled out by analysis and therefore, should have been considered).

- Fabrication restrictions to prohibit grinding were not applied to weld No. 5 (grinding promotes residual tensile stress state on ID of CRDM surface).

- Machining was performed on weld No. 5 during the fabrication process in order to achieve the dimensions and geometry specified in the design. This process induced cold work stresses in the weld.

- Material was changed from type 347 to type 316 stainless steel (both materials are essentially equally susceptible to TGSCC).

Also, in 1991, the Fort Calhoun plant had experienced through-wall leakage due to TGSCC at weld No. 5 of its CRDM housings (same housing design) and this operational experience had been reviewed by the licensee and dismissed. In the licensee's 2001 root cause evaluation, the licensee reviewed the weld build-up region failure by TGSCC at Fort Calhoun and concluded it would not occur at Palisades. This conclusion was based on the assumption that a higher oxygen environment (more aggressive environment) would exist in the Fort Calhoun housings than in the inservice Palisades housings. However the licensee did not confirm this assumption, nor did the licensee perform additional testing to determine if the environment of its inservice housings was sufficiently benign to prevent TGSCC. The licensee's 2012 RCT documented that due to organizational/ programmatic weakness at Palisades, the 1991 Fort Calhoun operating experience was not adequately utilized to include inspection of the weld No. 5. Similarly, the inspectors identified that the licensee had missed a key opportunity to implement effective corrective actions that could have prevented recurrence of the 2001 leakage event and had elected not to pursue that aspect further. Specifically, in EA-EAR-2001-0426-01 the licensee considered fabricating the replacement housings with Inconel 600 material because it was much more resistant to TGSCC, but ultimately decided not to do so. Additionally, various vendor reports were generated related to this issue in the mid 2000's. Those reports documented the potential susceptibility of weld No. 5 to TGSCC based upon a review of the CRDM housing conditions and available operating experience. The reports also noted that weld No. 5 was not inspected in any of the housings in 2001. One report in 2003 noted that weld No. 5 should have been examined as part of the action from the 2001 events since it was similar to Fort Calhoun. The issuance of these documents represented another opportunity for the licensee to identify the susceptibility of weld No. 5 to TGSCC prior to the cracking in CRDM-24.

The inspectors concluded the corrective actions taken in response to the 2001 CRDM through wall leak from TGSCC, a SCAQ, were not effective to preclude repetition. In particular, a through wall leak did recur on a CRDM from TGSCC. This issue was within the licensee's ability to foresee and correct; therefore, the issue was a performance deficiency. During the 2012 NRC special inspection, the NRC identified an URI for the TS pressure boundary leak. TS LCO 3.4.13 does not allow any primary coolant system

(PCS) pressure boundary leakage. In particular, TS Basis B3.4.13 "PCS Operational Leakage," explains that "No pressure boundary leakage from within the primary coolant pressure boundary is allowed, being indicative of material degradation. Leakage of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher leakage. Violation of this LCO could result in continued degradation of the primary coolant pressure boundary." Further, Action B, associated with this LCO, requires shutdown to mode 3 in 6 hours and mode 5 in 36 hours for such leakage. The licensee determined the CRDM-24 leakage commenced on or around July 14, 2012, and the plant continued to operate in this condition until August 12, 2012. Because the licensee was not aware of the existence of pressure boundary leakage, it failed to shut down the unit in six hours for a pressure boundary leak as required by TS 3.4.13 Action B. The NRC previously assessed the site's action for increasing unidentified leakage as part of the SIT. The NRC determined, at the time of higher unidentified leakage, the site took appropriate actions to attempt to locate the leak, eventually shutting down around .3 gallons per minute (gpm) leakage (earlier than the TS value of 1 gpm value for unidentified leakage). The licensee did not identify the source of the leakage as pressure boundary leakage until the shutdown on August 12, 2012, when a tour near the vessel head revealed the leaking housing. The pressure boundary leakage resulted in a TS violation and was due to the performance deficiency associated with the above mentioned Criterion XVI violation.

Based on the review discussed above, URIs 05000255/2012012-01 "TS for PCS Pressure Boundary Leakage" and 05000255/2012012-03 "Potential Failure to Take Corrective Actions to Prevent Recurrence of a Significant Condition Adverse to Quality" are closed.

Analysis: The inspectors determined that the licensee's failure to prevent recurrence of TGSCC of the CRDM housings (a SCAQ) that resulted in a violation of TS was a performance deficiency that warranted a significance evaluation. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone objective to limit the likelihood of events that upset plant stability. The issue was associated with the attribute of equipment performance. Specifically, the licensee did not take adequate corrective actions to prevent recurrence of leakage in CRDM housings, which represents pressure boundary leakage. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the slow rate of change for leakage for this cracking mechanism and this type of material. Type 316 stainless steel material under TGSCC will experience leakage rates well below a small break LOCA, which would be observed through the crack, alerting operators to take action to shut down the plant prior to experiencing a component rupture.

The cause of this finding, non-conservative decision making, occurred over ten years ago and is well outside of the nominal three year period in IMC 0612; and was not indicative of current performance, because no other opportunities to identify the issue occurred during the previous three year period. However more recently, the licensee exhibited non-conservative decision making with respect to addressing the potential for CRDM housing cracking and leakage during the recent root cause (Section 4OA2.3 (b.2) of this report), resulting in another finding. This cross-cutting aspect will be captured through the other finding.

Enforcement: During this inspection, the inspectors identified two NCVs of NRC requirements:

Title 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that, for significant conditions adverse to quality, the cause of the condition is determined and corrective action taken to preclude repetition.

TS LCO 3.4.13 requires PCS operational leakage be limited to "No pressure boundary LEAKAGE" when in Modes 1 through 4.

Contrary to the above, as of August 12, 2012, the licensee had failed to take corrective actions to preclude repetition for a SCAQ. Specifically, on June 21, 2001, the licensee discovered a through wall leak in CRDM-21 due to TGSCC and failed to reasonably include weld No. 5 in the corrective actions which resulted in a subsequent through wall leak in CRDM-24 due to TGSCC.

Contrary to the above, on or around July 14, 2012, PCS pressure boundary leakage at CRDM-24 existed while in Mode 1. Further, because the licensee was not aware that the leakage was PCS pressure boundary leakage, the licensee did not implement the associated TS action statement.

As a result of the second through wall leak, the licensee took corrective actions, which included the development of an inspection plan that would inspect weld No. 5 every outage until all CRDM housings were inspected.

Because these violations were of very low safety significance and were entered into the licensee's corrective action program as CR-PLP-2013-01134, these violations are being treated as an NCVs, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000255/2013002-*xx*; Failure to Take Corrective Action to Prevent Recurrence of CRDM Pressure Boundary Leakage).

.2 Failure to Adequately Address the Generic Implications of the Cracking identified in CRDM-24

Introduction: The inspectors identified a Finding with an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, for the licensee's failure to accomplish quality activities in accordance with the prescribed procedures. Specifically, the licensee failed to adequately evaluate and document the generic implications of the cause of the cracking identified in CRDM-24 in accordance with Procedure EN-LI-118, "Root Cause

Evaluation." This issue was entered into the licensee's corrective action program under CR-PLP-2013-05623.

Description: As a result of the cracking identified in CRDM-24, which was characterized as a SCAQ, the licensee performed a root cause evaluation in accordance with Procedure EN-LI-118. This procedure was identified as quality related and served to implement control pursuant to the licensee's quality assurance program. While reviewing the 2012 RCR (CR-PLP-2013-05623) related to the cracking identified in CRDM-24, the inspectors identified that the licensee had not appropriately considered the generic implications of the cracking in the extent of condition review. The licensee's proposed corrective actions, as a result of the 2012 RCR, narrowly focused on weld No. 5, instead of also including broader actions to ensure other CRDM housing welds were fit for their intended service life. These corrective actions consist of performing inspections of weld No. 5 on all CRDM housings.

On March 13, 2013, the inspectors requested that the licensee provide the bases for excluding other CRDM housing welds (weld No. 3 below weld No. 5 and weld No. 4 above weld No. 5) from the 2012 RCR scope of planned corrective actions. On March 29, 2013, the licensee provided additional information to justify excluding these welds from the scope of the corrective actions. The licensee credited the corrective actions associated with the modifications to the CRDM housing design completed in 2001 as the basis to exclude housing welds No. 3 and No. 4 from additional actions to identify the extent of TGSCC. The corrective actions taken in 2001 included performing heat sink welding, which is a methodology used to reduce the stresses on the inner ID of the weld. The licensee also changed the design to reduce design stresses at weld No. 3 and specified a smoother surface finish (RMS 125) to reduce potential crack initiation points. The licensee stated that these actions would produce compressive stresses on the ID of welds No. 3 and No. 4 making them immune from cracking. The inspectors acknowledged that these actions would reduce the tensile stress at the ID surface and thus reduce the probability of initiating TGSCC. However, the information provided did not demonstrate that TGSCC would not occur because it did not demonstrate that tensile stress would be eliminated at the ID surface during operation.

The inspectors identified that the three factors required for TGSCC could still be present at welds No. 3 and No. 4 as follows:

- Corrosive environment – Weld No. 3 would operate in a similar environment as weld No. 5 of the CRDM housing. Weld No. 4 would be exposed to a lower operating temperature than weld No. 5, however, TGSCC can still occur at 250 degrees Fahrenheit as evidenced by the Palisades previous operating experience with cracking identified in the seal housings that operate at even lower temperatures.

- Susceptible material – Welds No. 3 and No. 4 are composed of the same weld filler and base metal materials as weld No. 5 (e.g. weld filler material consistent with the type 316 stainless housing base metal). This material would be equally susceptible to TGSCC, as the type 347 stainless steel and weld filler materials used in the pre-2001 CRDM housing design that developed a through wall leak caused by TGSCC at weld No.3.

- Tensile stresses - While it is assumed that the corrective actions taken in response to the 2001 leak will reduce the potential for tensile stresses to exist on the inner surface of CRDM housings at welds No. 3 and No. 4, especially in light of repairs made to welds No. 3 and No. 4, it had not been conclusively demonstrated that these tensile stresses have been eliminated. As such, when evaluating welds No. 3 and No. 4 for applicability to the 2012 root cause, it was not reasonable to conclude that tensile stresses were not present, and therefore, the potential for TGSCC had been eliminated.

The 2012 RCR discussed manufacturing irregularities and misalignment between CRDM-24 and the support tube, seismic supports, and the associated reactor head penetration/CRDM nozzle as potential source of stresses leading to cracking. However, the RCR also stated that "based on the lack of cracking found in the other eight upper housings tested, the failed CRDM-24 upper housing contains an as-yet unidentified additional stress." Because the cause of the additional stress was not identified, the licensee had not established a basis in the RCR to exclude welds No. 3 and No. 4 from the extent of condition review (e.g. potential generic implications). In 2001, assumptions on crack growth rate and inspection intervals for welds No. 3 and No. 4 were made based on the information known at the time. The 2001 crack went through-wall after the CRDM was in service for 30 years and the cracking was widespread among the other CRDM housings. In 2012, the crack propagated through-wall after the CRDM was in service for 11 years and the cracking did not appear as widespread. Though TGSCC was a factor in both cracking events, there are still unknowns associated with the 2012 incident. The unknown additional stresses, as well as the time the CRDM was inservice before cracking in 2012, represent key differences as related to the cracking identified in 2001. In the 2012 RCR, the licensee did not consider these or other potential differences between the two incidents when determining not to include welds No. 3 and No. 4 in the evaluation and documentation of the generic implications of the root and contributing causes and therefore, did not provide a justification for excluding welds No. 3 and No.4 from this evaluation or corrective actions.

The inspectors identified that the licensee had not followed Procedure EN-LI-118, in the root cause review of the CRDM-24 leak as documented in report CR-PLP-2013-05623. Section 5.5 (12)e of EN-LI-118 required that the licensee "perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affect other SSCs." Additional details are provided in the procedure on how to conduct and document the evaluation. In this case, the inspectors identified that the licensee had not addressed or documented a basis in RCR CR-PLP-2013-05623 to exclude welds No. 3 and No. 4 from the generic factors discussed above that led to the 2012 leak in CRDM-24 (e.g. TGSCC at weld No. 5). The licensee entered this issue into the corrective action program as CR-PLP-2013-01500. Subsequently, the licensee decided to revise the inspection plan to add additional corrective actions to inspect a sample of welds No. 3 and No. 4 for TGSCC during the upcoming refueling outage.

Analysis: The inspectors determined that the failure to adequately evaluate and document the generic implications of the cause of the cracking identified in CRDM-24 in accordance with the root cause procedure EN-LI-118 was a performance deficiency that warranted a significance evaluation. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. The inspectors also answered "yes" to the More-than-Minor screening question, "if left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, absent NRC identification, the licensee would not have completed further evaluations or inspections of CRDM housing welds, which could have resulted in additional CRDM housing failure and leakage by TGSCC. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609, Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a large component rupture.

The inspectors determined that the primary cause of the failure to adequately consider welds No. 3 and No. 4 in the generic implications section of the RCR related to the cross-cutting component of Human Performance, Decision Making, because licensee staff did not use conservative assumptions in decision making. Specifically, the licensee did not use conservative assumptions when excluding welds No. 3 and No. 4 as being susceptible to TGSCC and therefore include them in the generic implications section of the RCR. (Item H.1(b) of IMC 310).

Enforcement: During the inspection, the inspectors identified one NCV of NRC requirements:

Title 10 CFR Part 50, Appendix B, Criterion V "Instructions, Procedures and Drawings requires in part, activities affecting quality shall be prescribed by documented procedures and shall be accomplished in accordance with these procedures.

Procedure EN-LI-118 "Root Cause Evaluation Process," Revision 17 states:

- 5.5 (12)e: perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affects other SSCs , organizations or work processes. Use the two step process in accordance with attachment 9.7
- Attachment 9.7: Determine whether the occurrence/consequence (problem) is isolated, or whether it has broader (generic or common mode) implications. Achieve this by asking the following questions:
    - i. Could this happen to equipment that is similar in function, design, or service condition?

    ii. Could this happen to a group of components? (components of the same construction or materials that could be similarly affected by one condition)

- Attachment 9.7: Document the results of the above considerations. Include the following items in the write up:
    i. Generic Implications (Is this problem/ cause limited to this component/equipment, or does it apply to others as well)
    ii. Existing broader (generic/common mode) considerations
- 5.5(15)(10)c&f: Document proposed corrective actions and due dates to address valid generic implications. If no corrective action is recommended for a valid generic implication then document the basis for this conclusion and any risk or consequence identified as a result of taking no action.

Contrary to the above, from February 24, 2013 through April 18, 2013, the licensee failed to accomplish activities affecting quality in accordance with procedure EN-LI-118, which was being implemented to correct a SCAQ. Specifically, the licensee failed to accomplish step 5.5 (12)e by not fully evaluating and documenting the existing broader (generic/common mode) considerations, extent of condition/cause associated with TGSCC at CRDM housing welds No. 3 and No. 4, including considering the susceptibility of the welds to TGSCC and the need to perform subsequent inspections or evaluations.

Subsequently, the licensee decided to revise the inspection plan to add additional corrective actions to inspect a sample of welds No. 3 and No. 4 for TGSCC during the upcoming refueling outage.

Because of the very low safety significance and because the licensee entered this issue into their corrective action program (CR-PLP-2013-01500), it is being treated as a NCV consistent with Section 2.3.2 of the Enforcement Policy (NCV 05000255/2013003-xx Failure to Adequately Address the Generic Implications of the Cracking Identified in CRDM-24).

4OA5 Other Activities

.1 (Closed) Unresolved Item 05000255/2012012-02: Potential Inadequate Degradation Evaluation of CRDM Housings (This inspection is part of the additional inspections included in the Palisades Deviation letter)

During a Special Inspection performed in August 2012, NRC inspectors identified an issue, which could not be resolved without additional information (URI). This issue was associated with the rate of growth of the crack which created the through wall leak in CRDM-24, discovered on August 12, 2012. Identification of this crack growth rate is significant in determining appropriate intervals for future inspections to provide reasonable assurance that CRDM housing leakage will not recur.

Preliminary failure analysis data available at the time of the inspection indicated that the observed cracking was due to TGSCC. Cracking of this type is normally due to the presence of oxygen and chlorides at the location of the crack. When examining the fracture surface at the location the through-wall leak occurred, the licensee identified six concentric rings (beach marks) propagating in a radial direction from the ID out towards the OD of the housing. Beach marks are normally associated with fatigue failures and indicate the number of stress cycles from crack initiation to crack failure. In this case, there was no evidence that fatigue contributed to the failure. Despite the lack of evidence of fatigue, it was apparent that the crack, which resulted in the CRDM-24 leak, grew in increments. It was not, however, immediately apparent whether the increments were related to oxygen ingress (refueling outages) or temperature/pressure cycles (heatups/cooldowns).

At the time of the original inspection, five time intervals for through wall crack growth were under consideration. Two were based on literature crack growth data and three were based on interpretations of the beach marks. These time intervals were:

- Based on literature data, one contractor estimated that a 10% through wall flaw would require four years to reach 50% through wall.
- Based on literature data another contractor estimated the crack growth rate to be 2.1 x 10^{-5} in/hr or 0.18 in/yr. This is approximately three times faster than the crack growth rate proposed in the above mentioned rate.
- Based on the concept of oxygen ingress at refueling outages six cycles of 18 months duration would require nine years for the crack to grow through wall
- Based on the concept of temperature/pressure cycles, the plant experienced six cold shutdowns in approximately two years preceding the crack. This equates to two years for the crack to grow through wall.
- Based on the concept that oxygen is required for crack growth and that oxygen is rapidly purged from the CRDM housings due to leakage past the seals, crack growth occurs only during the first few weeks of operation following a refueling outage, followed by no growth for the remaining period of operation when oxygen concentrations are low. This equates to six oxygen ingress events (irrespective of time between events) for the crack to grow through wall.

NRC inspectors including technical experts from NRC Headquarters performed a follow-up inspection to determine if the assumptions made by the licensee were conservative and the planned actions bounded those conservative assumptions. The inspectors reviewed a variety of documents associated with crack growth and inspection intervals. The inspectors noted the following statements included in the RCR and vendor documents related to the determination of the appropriate crack growth rate:

- The laboratory conducting the failure analysis concluded, it could not be conclusively determined if the beach marks corresponded to refueling outages, (i.e., 18 month cycle) or shorter periods as occurred during outages over the past 24 months
- Palisades CRDM-21 leaked at weld No. 3 in 2001. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 failure. In calculating the crack growth rate of this crack, one contractor utilized an interval between beach marks, which is much shorter than refueling outages. The intervals used are consistent with plant thermal cycles in which oxygen may or may not have been admitted into the CRDMs.
- A CRDM housing at Ft Calhoun leaked at weld No. 5 in 1990. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 Palisades failure. In calculating the crack growth rate of this crack, Ft Calhoun stated that the beach marks were related to refueling cycles. Ft Calhoun also performed calculations indicating that the oxygen level at the location of the flaw did

not change with time (including in response to refueling outages) because the CRDM housing was not vented. Ft Calhoun's evaluation indicated that oxygen levels at the vicinity of the crack would have begun to decline through diffusion and convection had the intervals between outages been much longer than 18 months. This is interpreted to mean that the beach marks at Ft Calhoun are in response to pressure/thermal cycles.

- In at least one instance, Palisades needed to repair the seals on a reactor coolant pump at a time other than an outage. This necessitated draining some of the water from the reactor coolant system and venting (admitting oxygen into) the CRDM housing. This represented an additional oxygen ingress event not included when determination of time to cracking is based on refueling outages.
- In its inspection plan, Palisades stated that it will inspect all CRDM housings over the next four refueling outages, i.e., the interval between inspections is one refueling outage

Based on the above review, the inspectors noted that there were certain non conservative statements contained in the RCR and the inspection plan. These included:

- The crack growth rate based on refueling outages was understated. If oxygen ingress is related to beach marks, given the oxygen ingress event which occurred to repair reactor coolant pump seals, six beach marks would occur in a maximum of five refueling intervals rather than the six refueling intervals that were used to calculate the crack growth rate in the RCR.
- The crack growth rate based on heat up and cool down cycles is overstated. The value in the root cause is based on 11 months. While six shutdowns did occur at the plant in 11 months several of these events did not result in pressure/temperature changes of the reactor coolant system. The appropriate timeframe is 24 months rather than 11.
- The inspection plan contains a non conservative statement: "However, once the crack has been initiated it propagates over four to five operating cycles prior to going through wall." While this statement does reflect one of the proposed theories for crack growth, sufficient evidence to demonstrate reasonable assurance that this theory is correct, and thereby overcome the non-conservatism of this statement, was not provided.

Despite the existence of the non conservatisms stated above, the inspectors concluded:

- Sufficient evidence to conclusively determine the rate of crack growth does not exist.
- Crack growth based on pressure/temperature cycles is the most conservative of the potential crack growth mechanisms. In the absence of reasonable assurance of the correctness of less conservative mechanisms, through wall crack growth in two years must be utilized for regulatory purposes.
- The licensee has not formally committed to any of the crack growth mechanisms discussed.
- The licensee's inspection program includes inspection of all of the CRDM housings over the next four refueling outages. Approximately 25% of the housings will be inspected during each outage. The inspection of 25% of the CRDM housings each interval is sufficient to indicate that, in the event no indications are found during a given inspection, that the probability that flaws exist in other housings is extremely low. As such, it may be considered that the inspection of approximately 25% of the

CRDM housings every refueling outage bounds all the crack growth rate mechanisms considered.

Overall, some weaknesses did exist in the site's assessment, but none of these issues arose above the level of a minor performance deficiency for the evaluations completed. With the corrective actions in place to monitor the CRDMs, the inspectors considered this approach to inspection to be both acceptable and sufficient justification to close this URI.

4OA6 Management Meetings

.2 Interim Exit Meetings

An interim exit was conducted for:

- The results of the selected issue follow-up inspection, with Mr. T. Vitali, Site Vice President on April 18, 2013.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

B. Davis, Engineering Director
O. Gustafson, Licensing Manager
T. Foudy, Engineering Supervisor
B. Williams, Engineer
B. Dotson, Licensing

LIST OF ITEMS OPENED, CLOSED, DISCUSSED

Closed

05000255/2012012-01	URI	TS for PCS Pressure Boundary Leakage
05000255/2012012-02	URI	Potential Inadequate Degradation Evaluation of CRDM Housings
05000255/2012012-03	URI	Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality

Opened and Discussed

None.

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

- EN-LI-118, Root Cause Evaluation Process, Revision 18
- SOP-1B, Primary Coolant System – Cooldown, Revision 15
WI0PCS-M-06, NSSS Walkdown, Revision 3
- C-PAL-01-02186, Root Cause Evaluation, Primary Coolant System Pressure Boundary Leakage CRD-21 Upper Housing Assembly
-CAP029079, Primary Coolant System Pressure Boundary Leakage CRD-21 Upper Housing, June 21, 2001
- CR-PLP-2012-05623, Root Cause Evaluation Report, CRD-24 Upper Housing Leak, Revision 2
- CR-PLP-2013-01500, PCRS Condition Summary (NRC identified Criterion V violation), April 3, 2013
CR-PLP-2013-01134, PCRS Condition Summary, (Criterion XVI Violation), March 15, 2013
- PLP-RPT-13-00007, Laboratory Analysis of Leaking CRDM #24 Housing from Palisades, Revision 0
- PLP-RPT-12-00123, Examination of Cracks in CRDM Housing #24, Revision 0
- PLP-RPT-13-00009, Summary of Technical Documents Addressing the CRDM Housing 24 cracking at the Palisades Nuclear Plant, Revision 0
- PLP-RPT-13-00006, CRDM Housing at the Palisades Nuclear Plant – Recommended Future Actions, Revision 0
- PLP-RPT-12-0012, Evaluation of Residual Stresses in Flaw in CRD Housing Weld Overlay – Palisades Nuclear Plant, Revision 0
- PLP-RPT-12-00121, Evaluation of Thermal Stresses at Flaw Location in CRD Upper Housing – Palisades Nuclear plant, Revision 0
- PLP-RPT-12-00128, Prior Evaluations of Palisades CRDM Housing, Revision 0
- PLP-RPT-12-00125, Leakage Calculation for CRDM Housing, Revision 0
- PLP-RPT-12-00124
- LPI Report A12315-LR-003, Evaluation of Inside Surface Stresses above Sub-surface Flaws at Flaw Location in CRDM #24 Upper Housing – Palisades Nuclear plant, Revision 0
- EA-EAR-2001-0373-04, Owner's Review of SI "Evaluation of Leakage from Circumferential and Axial Through-wall Cracks in Lower CRDM Housing", July 22, 2001
- EA-EAR-2001-0426-01, CRD Upper Housing Redesign, January 17, 2002
- EA-C-PAL-01-2186-02, CRD Upper Housing and Nozzle Weld Susceptibility Comparison, Revision 1
- ANP-2547NP, Transgranular Stress Corrosion Cracking of Austenitic Stainless Steels in CRDM Applications, Revision 1
- Project RP-1063, Supplier Verification Deficiency Reports, December 2001/January2002
- WPS 1149-3, Welding Procedure Specification (GTAW), Revision 3
- WCAP-16000, Review of the Root Cause Evaluation for Leakage from Palisades CRD-21 Upper Housing Assembly C-PAL-01-2186, October 2003

LIST OF ACRONYMS USED

CRDM	Control Rod Drive Mechanism
ID	Inside Diameter
GPM	Gallons per Minute
LOCA	Loss of Coolant Accident
NCV	Non-Cited Violation
NDE	Non-Destructive Examination
OD	Outer Diameter
PCS	Primary Coolant System
RCR	Root Cause Report
RCT	Root Cause Team
SCAQ	Significant Condition Adverse to Quality
SDP	Significance Determination Process
SIT	Special Inspection Team
TGSCC	Transgranular Stress Corrosion Cracking
TS	Technical Specification
URI	Unresolved Item
UT	Ultrasonic Examination

Hills, David

From: Sanchez Santiago, Elba
Sent: Monday, May 06, 2013 12:26 PM
To: Orth, Steven; Holmberg, Mel
Cc: Giessner, John; Hills, David
Subject: RE: Palisades Report Input- URI Closure CRDM-24

There were some slight editorial changes made. If you already reviewed Friday's version, there were no major changes made since then, therefore you wouldn't have to re-review this version.

-Elba

From: Orth, Steven
Sent: Monday, May 06, 2013 12:16 PM
To: Holmberg, Mel
Cc: Sanchez Santiago, Elba; Giessner, John; Hills, David
Subject: RE: Palisades Report Input- URI Closure CRDM-24

Is this different from Friday's version?

From: Holmberg, Mel
Sent: Monday, May 06, 2013 11:12 AM
To: Orth, Steven
Cc: Sanchez Santiago, Elba; Giessner, John; Hills, David
Subject: Palisades Report Input- URI Closure CRDM-24

Steve,

The attached is the most recent version of the Palisades report with the violations proposed associated with the CRDM-24 housing leaks. This version has all of the review comments including yours incorporated. If you have any further comments please let us know today if possible,

Thanks,

Mel

C4

UNITED STATES
NUCLEAR REGULATORY COMMISSION
LISLE, IL 60532-4352

May XX, 2012

MEMORANDUM TO: Thomas Taylor
Senior Resident Inspector
Palisades Nuclear Plant

FROM: David Hills, Chief
Engineering Branch 3
Division of Reactor Safety

SUBJECT: PALISADES NUCLEAR PLANT
DRS INPUT TO INTEGRATED REPORT 05000255/2013002

Enclosed is the report input for the Palisades Nuclear Plant, Inspection Report 05000255/2013002. This report input documents completion of our review of Unresolved Items 05000255/2012012-01, "TS for PCS Pressure Boundary Leakage," 05000255/2012012-02, "Potential Inadequate Degradation Evaluation of CRDM Housings," and 05000255/2012012-03, "Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality." This report also completes one sample of the Problem Identification and Resolution, Selected Issue Follow-up inspection in accordance with IP 71152. I have reviewed this input to confirm compliance with Inspection Manual Chapter (IMC) 0612 and IMC 0305. This input is ready for inclusion into the integrated report and dissemination to the public.

Please input the following post Inspection Data into RPS:

Inspection Procedure	**Procedure Status** – see below: Incomplete, Complete, Complete by reference, Complete-full sample not available, Complete – opportunity to apply procedure not available, Not Applicable.	**Sample Size** – As documented in Scope Section If less than full sample size documented in the report input, the inspector must provide a justification below to enter into RPS and support the procedure status selected
71152	Complete	1

Inspection Report Item and Type (AV, FIN, NCV, URI or VIO)	**Cornerstone** (IE, MS, BI, EP, OR, PR, MISC)	**Cross Cutting Aspect** (H.n(i), P.n(i), S.n(i))	**Responsible Person/Owner**	**Procedure or TI** (71111.07T)	**RPS Branch Code (e.g. closeout responsibility)** EB1 3820 EB2 3870 EB3 3840 PST (RP) 3860 PSB (Safeguards) 3850 OB 3810
NCV-XXX	IE	n/a	E. Sanchez Santiago	71152	3820
NCV-XXX	IE	H.1(b)	E. Sanchez Santiago	71152	3820

Enclosure: Input to Inspection Report 05000255/2013002

cc w/encl: J. Giessner, Chief
C. Hernandez, Site Admin Assistant

CONTACT: E. Sanchez Santiago, DRS
(630) 829-9715

DOCUMENT NAME: G:\DRSIII\DRS\Work in Progress\-Palisades Input to DRP Report 2013 002 URI EMS.docx
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OFFICE	RIII		NRR		RIII		NRR	
NAME	ESanchezSantiago		DAlley		DHills		TLupold	
DATE	5/ /13							

OFFICIAL RECORD COPY

Cover Letter

_X_ Green findings involving a violation were identified. Include the following:

Based on the results of this inspection, two NRC-identified findings of very low safety significance (Green) were identified. These findings were determined to involve a violation of NRC requirements. However, because of the very low safety significance and because the issues were entered into your corrective action program, the NRC is treating the issue as Non-Cited Violation, in accordance with Section 2.3.2 of the NRC Enforcement Policy.

TITLE PAGE

Inspectors: D. Alley, Senior Materials Engineer
E. Sanchez Santiago, Reactor Inspector

SUMMARY OF FINDINGS

A. <u>NRC-Identified and Self-Revealed Findings</u>

Cornerstones: Initiating Events

- <u>Green.</u> A self-revealing Green Finding with associated Non-Cited Violations (NCV) of 10 CFR Part 50, Appendix B, Criterion XVI and Technical Specification (TS) 3.4.13 Primary Coolant System (PCS) Operational Leakage, was identified for failure to take corrective actions to prevent recurrence of Control Rod Drive Mechanism (CRDM) cracking and leakage, a significant condition adverse to quality (SCAQ), and resulting in operation of the reactor with PCS pressure boundary leakage. Specifically, for Criterion XVI the licensee failed to include the internal CRDM housing weld build-up area within the scope of corrective actions taken for a 2001 CRDM through wall leak on CRDM-21, caused by transgranular stress corrosion cracking (TGSCC). Subsequently, a through wall leak recurred in the weld build-up area on CRDM-24 in 2012 due to TGSCC. As a result, the licensee operated with PCS pressure boundary leakage, which is not allowed by TS 3.4.13. Further, because the licensee was not aware that the leakage was PCS pressure boundary leakage, the licensee did not implement the associated TS action statement. The licensee replaced CRDM-24 upper housing and wrote CR-PLP-2013-01134. Additional corrective actions are described in NRC Inspection Report 05000255/2012012.

  The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone objective to limit the likelihood of events that upset plant stability. The issue was associated with the attribute of equipment performance. Specifically, the licensee did not take adequate corrective actions to prevent recurrence of leakage in CRDM housings, which represents pressure boundary leakage. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System Loss of Coolant Accident (LOCA) initiator contributor. The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the slow rate of change for leakage for this cracking mechanism and this type of material. Type 316 stainless steel material under TGSCC will experience leakage rates well below a small break LOCA, which would be observed through the crack, alerting operators to take action to shut down the plant prior to experiencing a component rupture. The cause of this finding, non-conservative decision making, occurred over ten years ago and is well outside of the nominal three year period in IMC 0612; and would not be indicative of current performance, unless there were other opportunities to identify the issue; therefore, the inspectors concluded this was not indicative of current performance. However more recently, the licensee exhibited non-conservative decision making with respect to addressing the potential for CRDM housing cracking and leakage during the recent root cause (Section 4OA2.3 (b.2) of this report), resulting in another finding. This cross-cutting aspect will be captured through the other finding. (Section 4OA2.3(b.1))

- Green. The inspectors identified a Finding with an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, for the licensee's failure to accomplish quality activities in accordance with the prescribed procedures. Specifically, the licensee failed to adequately evaluate and document the generic implications of the cause of the 2012 cracking identified in CRDM-24 in accordance with Procedure EN-LI-118 "Root Cause Evaluation." This issue was entered into the licensee's corrective action program under CR-PLP-2013-01500. Subsequently, the licensee decided to revise the inspection plan to add additional corrective actions to inspect a sample of welds No. 2 and No. 4 for transgranular stess corrosion cracking during the upcoming refueling outage.

  The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because the inspectors answered "yes" to the More-than-Minor screening question, "if left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, absent NRC identification, the licensee would not have completed further evaluations or inspections of CRDM housing welds, which could have resulted in additional CRDM housing failure and leakage by TGSCC. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System LOCA initiator contributor. The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609, Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a

LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the slow rate of change for leakage for this cracking mechanism and this type of material. Type 316 stainless steel material under TGSCC will experience leakage rates well below a small break LOCA, which would be observed through the crack, alerting operators to take action to shut down the plant prior to experiencing a component rupture. The inspectors determined that the primary cause of the failure to adequately consider welds No. 3 and No. 4 in the generic implications section of the root cause report (RCR) related to the cross-cutting component of Human Performance, Decision Making, because licensee staff did not use conservative assumptions in decision making. Specifically, the licensee did not use conservative assumptions when excluding welds No. 3 and No. 4 as being susceptible to TGSCC when there was not enough information to exclude them from consideration. (Item H.1(b)). (Section 4OA2.3(b.2))

B. Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAILS

4. REACTOR SAFETY

4OA2 Identification and Resolution of Problems (71152)

.3 Selected Issue Follow-up Inspection: Through Wall Leakage of CRDM-24 (This inspection is part of the additional inspections referenced in the Palisades Deviation letter.)

a. Inspection Scope

On August 12, 2012, the licensee shut down the plant to investigate an increase in unidentified leakage. The source of the leakage was determined to be a crack in CRDM-24. The NRC dispatched a special inspection team (SIT) to review the CRDM-24 leakage event. The results of that inspection are provided in Inspection Report 05000255/2012012. The licensee completed an evaluation to determine the cause of the cracking (CR-PLP-2012-05623).

From March 4, 2013 to March 15, 2013, the inspectors completed one inspection sample regarding problem identification and resolution based upon review of the licensee's RCR contained in corrective action document CR-PLP-2012-05623. In addition, the inspectors performed reviews related to three Unresolved Items (URI) identified during the SIT inspection:

- URI 05000255/2012012-01 Technical Specification (TS) for PCS Pressure Boundary Leakage. (The closure of this URI is documented in section 4OA2.3 (b.1) of this report.)

- URI 05000255/2012012-02 Potential Inadequate Degradation Evaluation of CRDM Housings (The closure of this URI is documented in section 4OA5.1 of this report.)

- URI 05000255/2012012-03 Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality (The closure of this URI is documented in section 4OA2.3 (b.1) of this report.)

The inspectors reviewed the licensee's actions in accordance with performance attributes identified in IP 71152. Specifically, the inspectors reviewed licensee corrective action records to determine if: (1) the problems were accurately identified; (2) operability and reportability were adequately ascertained; (3) extent of condition and generic implications were appropriately addressed; (4) classification and prioritization of the problem were commensurate with safety significance; (5) root and contributing causes were identified; (6) corrective actions were appropriately focused to correct the problem; and (7) timely corrective actions were completed or proposed commensurate with the safety significance of the issues.

b. Findings

.1 Failure to Take Corrective Actions to Prevent Recurrence of CRDM Housing Cracking and Leakage

Introduction: A self-revealing Green Finding with associated NCV of 10 CFR Part 50, Appendix B, Criterion XVI and TS 3.4.13 PCS Operational Leakage, was identified for failure to take corrective actions to prevent recurrence of CRDM cracking and leakage, a SCAQ, and resulting in operation of the reactor with PCS pressure boundary leakage. Specifically, for Criterion XVI the licensee failed to include the internal CRDM housing weld build-up area within the scope of corrective actions taken for a 2001 CRDM through wall leak on CRDM-21 caused by TGSCC. Subsequently, a through wall leak recurred in the weld build-up area on CRDM-24 in 2012 due to TGSCC. As a result, the licensee operated with PCS pressure boundary leakage, which is not allowed by TS 3.4.13. Further, because the licensee was not aware that the leakage was PCS pressure boundary leakage, the licensee did not implement the associated TS action statement.

Description: In 2001, the licensee discovered a steam leak in the housing of CRDM-21 caused by a through-wall TGSCC at CRDM housing weld No. 3, which was located just below the weld build-up region (weld No. 5). Weld No. 5 consists of a weld material deposit applied to the inside diameter (ID) of the CRDM housing which provides for alignment of the CRDM. This issue was categorized as a SCAQ by the licensee (CPAL0102186) because it represented a break in the reactor system pressure boundary. The licensee's root cause evaluation was documented in RCR/C-PAL-01-02186 and concluded that the cracks in CRDM-21 were caused by TGSCC, which occurred in areas of heavy grinding or machining tool marks. Specifically, this leak was the result of an ID initiated, axially oriented, transgranular crack in the austenitic stainless steel housing material. The failure analysis performed in response to this event identified both axial and circumferential cracks associated with weld No. 3. Extent of condition inspections revealed additional, non-through wall cracks associated with weld No. 3 in 41 of the 44 remaining housings for a total of 42 of 45 housings containing cracks.

In response to the 2001 cracking, Palisades replaced all 45 CRDM housings with housings thought to be more resistant to cracking. Principle changes included:

- Elimination of weld No. 2,
- Relocation of weld No. 3 to a higher location thereby minimizing the deposition of crud in the gap between the weld and the bottom plate of the rack and pinion assembly,
- Reduction in residual stresses and cold work on welds by requiring better surface finishes, and
- Use of heat sink welding to reduce ID residual tensile stresses.

In January of 2002, an NRC SIT (reference IR 50-2555/01-15) reviewed the licensee proposed corrective actions associated with the through-wall leakage of the CRDM-21 housing caused by TGSCC. The 2001 RCR reviewed by the NRC stated the action to prevent recurrence was to "develop and implement an inspection plan to address areas and components identified in Attachment C-Extent of Condition. One of the components included in Attachment C was the CRDM. The recommended action was to perform volumetric inspection of the welds contained in the CRDM. Subsequently, the licensee decided to change exclude weld No. 5.

Following the subsequent 2012 CRDM-24 leak, the licensee determined the leak occurred because of a through-wall crack adjacent to weld No. 5. The licensee formed a root cause team (RCT) staffed with licensee personnel and augmented with input from vendors. The root cause investigation was conducted in accordance with site procedure EN-LI-118 "Root Cause Evaluation Process" and was documented in root cause analysis report CR-PLP-2012-05623. In this report, the licensee's RCT determined that the probable cause of the cracking was:

> "Stresses in the weld build up area due to manufacturing irregularities and misalignments between CRDM-24 upper housing, support tube, and the associated reactor head penetration/CRDM nozzle. Based on lack of cracking found in the other eight upper housings tested, the failed CRDM-24 upper housing contains an as-yet unidentified additional stress."

The RCT also identified the following contributing cause:

> "TGSCC initiating within the internal weld build-up material of CRDM-24. The through wall crack initiated in the weld material and then propagated through the base metal until a leak developed in the outer diameter (OD) witness band region at the base of the ID weld build up.

This conclusion was based upon destructive and non destructive examinations (NDE) completed on a section of the failed housing, which included the through-wall flaw. The RCT also relied upon vendor technical reports assessing the results of the NDE as well as vendor calculations related to the stresses in the CRDM housings.

To determine the extent of condition, the licensee performed ultrasonic (UT) examinations of weld No. 5 on eight additional CRDM housings. The licensee selected these housings based on being in a similar location on the head as CRDM-24, and previous cracking having been identified in some of these housings prior to the replacement of the CRDM upper housings and seal housings in 2002. The inspectors concluded that this was an adequate sample for an initial extent of condition review based upon the concept that, in light of eight negative exams, the statistical probability of a flaw in the remaining CRDM housings was very low. Additionally, the licensee planned to conduct examinations of more housings during the next refueling outage.

The inspectors concluded that the licensee actions following the 2001 leak were not adequate because the appropriate actions to preclude recurrence were within the licensee's ability to foresee and implement. Specifically, the inspectors concluded that the licensee did not effectively implement corrective actions for the 2001 CRDM housing leak resulting in the 2012 CRDM-24 housing leak.

Licensee corrective actions taken in response to the 2001 event were limited to butt welds. The inspectors reviewed the licensee actions to determine if they had been sufficient to eliminate one of the three necessary factors to cause TGSCC on the CRDM housings: (1) a susceptible material, (2) a corrosive environment and (3) tensile stress. The inspectors identified that the licensee had failed to eliminate one or more of the necessary factors at weld No. 5 (which was not a butt weld) to preclude TGSCC in the replacement housing. Specifically:

- The licensee's 2001 RCR documented that weld No. 5 is exposed to essentially the same environment as the weld that experienced the cracking (corrosive environment remained unchanged).

- No analysis was completed on the stress conditions for weld No. 5 prior to approving the modified replacement housing design (the potential for residual tensile weld stresses on ID of CRDM surface was not ruled out by analysis and therefore, should have been considered).

- Fabrication restrictions to prohibit grinding were not applied to weld No. 5 (grinding promotes residual tensile stress state on ID of CRDM surface).

- Machining was performed on weld No. 5 during the fabrication process in order to achieve the dimensions and geometry specified in the design. This process induced cold work stresses in the weld.

- Material was changed from type 347 to type 316 stainless steel (both materials are essentially equally susceptible to TGSCC).

Also, in 1991, the Fort Calhoun plant had experienced through-wall leakage due to TGSCC at weld No. 5 of its CRDM housings (same housing design) and this operational experience had been reviewed by the licensee and dismissed. In the licensee's 2001 root cause evaluation, the licensee reviewed the weld build-up region failure by TGSCC at Fort Calhoun and concluded it would not occur at Palisades. This conclusion was based on the assumption that a higher oxygen environment (more aggressive environment) would exist in the Fort Calhoun housings than in the inservice Palisades housings. However the licensee did not confirm this assumption, nor did the licensee perform additional testing to determine if the environment of its inservice housings was sufficiently benign to prevent TGSCC. The licensee's 2012 RCT documented that due to organizational/ programmatic weakness at Palisades, the 1991 Fort Calhoun operating experience was not adequately utilized to include inspection of the weld No. 5. Similarly, the inspectors identified that the licensee had missed a key opportunity to implement effective corrective actions that could have prevented recurrence of the 2001 leakage event and had elected not to pursue that aspect further. Specifically, in EA-EAR-2001-0426-01 the licensee considered fabricating the replacement housings with Inconel 600 material because it was much more resistant to TGSCC, but ultimately decided not to do so. Additionally, various vendor reports were generated related to this issue in the mid 2000's. Those reports documented the potential susceptibility of weld No. 5 to TGSCC based upon a review of the CRDM housing conditions and available operating experience. The reports also noted that weld No. 5 was not inspected in any of the housings in 2001. One report in 2003 noted that weld No. 5 should have been examined as part of the action from the 2001 events since it was similar to Fort Calhoun. The issuance of these documents represented another opportunity for the licensee to identify the susceptibility of weld No. 5 to TGSCC prior to the cracking in CRDM-24.

The inspectors concluded the corrective actions taken in response to the 2001 CRDM through wall leak from TGSCC, a SCAQ, were not effective to preclude repetition. In particular, a through wall leak did recur on a CRDM from TGSCC. This issue was within the licensee's ability to foresee and correct; therefore, the issue was a performance deficiency. During the 2012 NRC special inspection, the NRC identified an URI for the TS pressure boundary leak. TS LCO 3.4.13 does not allow any primary coolant system

(PCS) pressure boundary leakage. In particular, TS Basis B3.4.13 "PCS Operational Leakage," explains that "No pressure boundary leakage from within the primary coolant pressure boundary is allowed, being indicative of material degradation. Leakage of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher leakage. Violation of this LCO could result in continued degradation of the primary coolant pressure boundary." Further, Action B, associated with this LCO, requires shutdown to mode 3 in 6 hours and mode 5 in 36 hours for such leakage. The licensee determined the CRDM-24 leakage commenced on or around July 14, 2012, and the plant continued to operate in this condition until August 12, 2012. Because the licensee was not aware of the existence of pressure boundary leakage, it failed to shut down the unit in six hours for a pressure boundary leak as required by TS 3.4.13 Action B. The NRC previously assessed the site's action for increasing unidentified leakage as part of the SIT. The NRC determined, at the time of higher unidentified leakage, the site took appropriate actions to attempt to locate the leak, eventually shutting down around .3 gallons per minute (gpm) leakage (earlier than the TS value of 1 gpm value for unidentified leakage). The licensee did not identify the source of the leakage as pressure boundary leakage until the shutdown on August 12, 2012, when a tour near the vessel head revealed the leaking housing. The pressure boundary leakage resulted in a TS violation and was due to the performance deficiency associated with the above mentioned Criterion XVI violation.

Based on the review discussed above, URIs 05000255/2012012-01 "TS for PCS Pressure Boundary Leakage" and 05000255/2012012-03 "Potential Failure to Take Corrective Actions to Prevent Recurrence of a Significant Condition Adverse to Quality" are closed.

Analysis: The inspectors determined that the licensee's failure to prevent recurrence of TGSCC of the CRDM housings (a SCAQ) that resulted in a violation of TS was a performance deficiency that warranted a significance evaluation. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone objective to limit the likelihood of events that upset plant stability. The issue was associated with the attribute of equipment performance. Specifically, the licensee did not take adequate corrective actions to prevent recurrence of leakage in CRDM housings, which represents pressure boundary leakage. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609 Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the slow rate of change for leakage for this cracking mechanism and this type of material. Type 316 stainless steel material under TGSCC will experience leakage rates well below a small break LOCA, which would be observed through the crack, alerting operators to take action to shut down the plant prior to experiencing a component rupture.

The cause of this finding, non-conservative decision making, occurred over ten years ago and is well outside of the nominal three year period in IMC 0612; and was not indicative of current performance, because no other opportunities to identify the issue occurred during the previous three year period. However more recently, the licensee exhibited non-conservative decision making with respect to addressing the potential for CRDM housing cracking and leakage during the recent root cause (Section 4OA2.3 (b.2) of this report), resulting in another finding. This cross-cutting aspect will be captured through the other finding.

Enforcement: During this inspection, the inspectors identified two NCVs of NRC requirements:

Title 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that, for significant conditions adverse to quality, the cause of the condition is determined and corrective action taken to preclude repetition.

TS LCO 3.4.13 requires PCS operational leakage be limited to "No pressure boundary LEAKAGE" when in Modes 1 through 4.

Contrary to the above, as of August 12, 2012, the licensee had failed to take corrective actions to preclude repetition for a SCAQ. Specifically, on June 21, 2001, the licensee discovered a through wall leak in CRDM-21 due to TGSCC and failed to reasonably include weld No. 5 in the corrective actions which resulted in a subsequent through wall leak in CRDM-24 due to TGSCC.

Contrary to the above, on or around July 14, 2012, PCS pressure boundary leakage at CRDM-24 existed while in Mode 1. Further, because the licensee was not aware that the leakage was PCS pressure boundary leakage, the licensee did not implement the associated TS action statement.

As a result of the second through wall leak, the licensee took corrective actions, which included the development of an inspection plan that would inspect weld No. 5 every outage until all CRDM housings were inspected.

Because these violations were of very low safety significance and were entered into the licensee's corrective action program as CR-PLP-2013-01134, these violations are being treated as an NCVs, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000255/2013002-*XX*; Failure to Take Corrective Action to Prevent Recurrence of CRDM Pressure Boundary Leakage).

.2 Failure to Adequately Address the Generic Implications of the Cracking identified in CRDM-24

Introduction: The inspectors identified a Finding with an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, for the licensee's failure to accomplish quality activities in accordance with the prescribed procedures. Specifically, the licensee failed to adequately evaluate and document the generic implications of the cause of the cracking identified in CRDM-24 in accordance with Procedure EN-LI-118, "Root Cause

Evaluation." This issue was entered into the licensee's corrective action program under CR-PLP-2013-05623.

Description: As a result of the cracking identified in CRDM-24, which was characterized as a SCAQ, the licensee performed a root cause evaluation in accordance with Procedure EN-LI-118. This procedure was identified as quality related and served to implement control pursuant to the licensee's quality assurance program. While reviewing the 2012 RCR (CR-PLP-2013-05623) related to the cracking identified in CRDM-24, the inspectors identified that the licensee had not appropriately considered the generic implications of the cracking in the extent of condition review. The licensee's proposed corrective actions, as a result of the 2012 RCR, narrowly focused on weld No. 5, instead of also including broader actions to ensure other CRDM housing welds were fit for their intended service life. These corrective actions consist of performing inspections of weld No. 5 on all CRDM housings.

On March 13, 2013, the inspectors requested that the licensee provide the bases for excluding other CRDM housing welds (weld No. 3 below weld No. 5 and weld No. 4 above weld No. 5) from the 2012 RCR scope of planned corrective actions. On March 29, 2013, the licensee provided additional information to justify excluding these welds from the scope of the corrective actions. The licensee credited the corrective actions associated with the modifications to the CRDM housing design completed in 2001 as the basis to exclude housing welds No. 3 and No. 4 from additional actions to identify the extent of TGSCC. The corrective actions taken in 2001 included performing heat sink welding, which is a methodology used to reduce the stresses on the inner ID of the weld. The licensee also changed the design to reduce design stresses at weld No. 3 and specified a smoother surface finish (RMS 125) to reduce potential crack initiation points. The licensee stated that these actions would produce compressive stresses on the ID of welds No. 3 and No. 4 making them immune from cracking. The inspectors acknowledged that these actions would reduce the tensile stress at the ID surface and thus reduce the probability of initiating TGSCC. However, the information provided did not demonstrate that TGSCC would not occur because it did not demonstrate that tensile stress would be eliminated at the ID surface during operation.

The inspectors identified that the three factors required for TGSCC could still be present at welds No. 3 and No. 4 as follows:

- Corrosive environment – Weld No. 3 would operate in a similar environment as weld No. 5 of the CRDM housing. Weld No. 4 would be exposed to a lower operating temperature than weld No. 5, however, TGSCC can still occur at 250 degrees Fahrenheit as evidenced by the Palisades previous operating experience with cracking identified in the seal housings that operate at even lower temperatures.

- Susceptible material – Welds No. 3 and No. 4 are composed of the same weld filler and base metal materials as weld No. 5 (e.g. weld filler material consistent with the type 316 stainless housing base metal). This material would be equally susceptible to TGSCC, as the type 347 stainless steel and weld filler materials used in the pre-2001 CRDM housing design that developed a through wall leak caused by TGSCC at weld No.3.

- Tensile stresses - While it is assumed that the corrective actions taken in response to the 2001 leak will reduce the potential for tensile stresses to exist on the inner surface of CRDM housings at welds No. 3 and No. 4, especially in light of repairs made to welds No. 3 and No. 4, it had not been conclusively demonstrated that these tensile stresses have been eliminated. As such, when evaluating welds No. 3 and No. 4 for applicability to the 2012 root cause, it was not reasonable to conclude that tensile stresses were not present, and therefore, the potential for TGSCC had been eliminated.

The 2012 RCR discussed manufacturing irregularities and misalignment between CRDM-24 and the support tube, seismic supports, and the associated reactor head penetration/CRDM nozzle as potential source of stresses leading to cracking. However, the RCR also stated that "based on the lack of cracking found in the other eight upper housings tested, the failed CRDM-24 upper housing contains an as-yet unidentified additional stress." Because the cause of the additional stress was not identified, the licensee had not established a basis in the RCR to exclude welds No. 3 and No. 4 from the extent of condition review (e.g. potential generic implications). In 2001, assumptions on crack growth rate and inspection intervals for welds No. 3 and No. 4 were made based on the information known at the time. The 2001 crack went through-wall after the CRDM was in service for 30 years and the cracking was widespread among the other CRDM housings. In 2012, the crack propagated through-wall after the CRDM was in service for 11 years and the cracking did not appear as widespread. Though TGSCC was a factor in both cracking events, there are still unknowns associated with the 2012 incident. The unknown additional stresses, as well as the time the CRDM was inservice before cracking in 2012, represent key differences as related to the cracking identified in 2001. In the 2012 RCR, the licensee did not consider these or other potential differences between the two incidents when determining not to include welds No. 3 and No. 4 in the evaluation and documentation of the generic implications of the root and contributing causes and therefore, did not provide a justification for excluding welds No. 3 and No.4 from this evaluation or corrective actions.

The inspectors identified that the licensee had not followed Procedure EN-LI-118, in the root cause review of the CRDM-24 leak as documented in report CR-PLP-2013-05623. Section 5.5 (12)e of EN-LI-118 required that the licensee "perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affect other SSCs." Additional details are provided in the procedure on how to conduct and document the evaluation. In this case, the inspectors identified that the licensee had not addressed or documented a basis in RCR CR-PLP-2013-05623 to exclude welds No. 3 and No. 4 from the generic factors discussed above that led to the 2012 leak in CRDM-24 (e.g. TGSCC at weld No. 5). The licensee entered this issue into the corrective action program as CR-PLP-2013-01500. Subsequently, the licensee decided to revise the inspection plan to add additional corrective actions to inspect a sample of welds No. 3 and No. 4 for TGSCC during the upcoming refueling outage.

Analysis: The inspectors determined that the failure to adequately evaluate and document the generic implications of the cause of the cracking identified in CRDM-24 in accordance with the root cause procedure EN-LI-118 was a performance deficiency that warranted a significance evaluation. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. The inspectors also answered "yes" to the More-than-Minor screening question, "if left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?" Specifically, absent NRC identification, the licensee would not have completed further evaluations or inspections of CRDM housing welds, which could have resulted in additional CRDM housing failure and leakage by TGSCC. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness" of IMC 609, Attachment 4 "Initial Characterization of Findings" issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609, Attachment A "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g. flaw tolerance) of the type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a large component rupture.

The inspectors determined that the primary cause of the failure to adequately consider welds No. 3 and No. 4 in the generic implications section of the RCR related to the cross-cutting component of Human Performance, Decision Making, because licensee staff did not use conservative assumptions in decision making. Specifically, the licensee did not use conservative assumptions when excluding welds No. 3 and No. 4 as being susceptible to TGSCC and therefore include them in the generic implications section of the RCR. (Item H.1(b) of IMC 310).

Enforcement: During the inspection, the inspectors identified one NCV of NRC requirements:

Title 10 CFR Part 50, Appendix B, Criterion V "Instructions, Procedures and Drawings requires in part, activities affecting quality shall be prescribed by documented procedures and shall be accomplished in accordance with these procedures.

Procedure EN-LI-118 "Root Cause Evaluation Process," Revision 17 states:

- 5.5 (12)e: perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affects other SSCs , organizations or work processes. Use the two step process in accordance with attachment 9.7
- Attachment 9.7: Determine whether the occurrence/consequence (problem) is isolated, or whether it has broader (generic or common mode) implications. Achieve this by asking the following questions:
    i. Could this happen to equipment that is similar in function, design, or service condition?

    ii. Could this happen to a group of components? (components of the same construction or materials that could be similarly affected by one condition)

- Attachment 9.7: Document the results of the above considerations. Include the following items in the write up:
    i. Generic Implications (Is this problem/ cause limited to this component/equipment, or does it apply to others as well)
    ii. Existing broader (generic/common mode) considerations
- 5.5(15)(10)c&f: Document proposed corrective actions and due dates to address valid generic implications. If no corrective action is recommended for a valid generic implication then document the basis for this conclusion and any risk or consequence identified as a result of taking no action.

Contrary to the above, from February 24, 2013 through April 18, 2013, the licensee failed to accomplish activities affecting quality in accordance with procedure EN-LI-118, which was being implemented to correct a SCAQ. Specifically, the licensee failed to accomplish step 5.5 (12)e by not fully evaluating and documenting the existing broader (generic/common mode) considerations, extent of condition/cause associated with TGSCC at CRDM housing welds No. 3 and No. 4, including considering the susceptibility of the welds to TGSCC and the need to perform subsequent inspections or evaluations.

Subsequently, the licensee decided to revise the inspection plan to add additional corrective actions to inspect a sample of welds No. 3 and No. 4 for TGSCC during the upcoming refueling outage.

Because of the very low safety significance and because the licensee entered this issue into their corrective action program (CR-PLP-2013-01500), it is being treated as a NCV consistent with Section 2.3.2 of the Enforcement Policy (NCV 05000255/2013003-xx Failure to Adequately Address the Generic Implications of the Cracking Identified in CRDM-24).

4OA5 <u>Other Activities</u>

.1 <u>(Closed) Unresolved Item 05000255/2012012-02: Potential Inadequate Degradation Evaluation of CRDM Housings (This inspection is part of the additional inspections included in the Palisades Deviation letter)</u>

During a Special Inspection performed in August 2012, NRC inspectors identified an issue, which could not be resolved without additional information (URI). This issue was associated with the rate of growth of the crack which created the through wall leak in CRDM-24, discovered on August 12, 2012. Identification of this crack growth rate is significant in determining appropriate intervals for future inspections to provide reasonable assurance that CRDM housing leakage will not recur.

Preliminary failure analysis data available at the time of the inspection indicated that the observed cracking was due to TGSCC. Cracking of this type is normally due to the presence of oxygen and chlorides at the location of the crack. When examining the fracture surface at the location the through-wall leak occurred, the licensee identified six concentric rings (beach marks) propagating in a radial direction from the ID out towards the OD of the housing. Beach marks are normally associated with fatigue failures and indicate the number of stress cycles from crack initiation to crack failure. In this case, there was no evidence that fatigue contributed to the failure. Despite the lack of evidence of fatigue, it was apparent that the crack, which resulted in the CRDM-24 leak, grew in increments. It was not, however, immediately apparent whether the increments were related to oxygen ingress (refueling outages) or temperature/pressure cycles (heatups/cooldowns).

At the time of the original inspection, five time intervals for through wall crack growth were under consideration. Two were based on literature crack growth data and three were based on interpretations of the beach marks. These time intervals were:

- Based on literature data, one contractor estimated that a 10% through wall flaw would require four years to reach 50% through wall.
- Based on literature data another contractor estimated the crack growth rate to be 2.1 x 10^{-5} in/hr or 0.18 in/yr. This is approximately three times faster than the crack growth rate proposed in the above mentioned rate.
- Based on the concept of oxygen ingress at refueling outages six cycles of 18 months duration would require nine years for the crack to grow through wall
- Based on the concept of temperature/pressure cycles, the plant experienced six cold shutdowns in approximately two years preceding the crack. This equates to two years for the crack to grow through wall.
- Based on the concept that oxygen is required for crack growth and that oxygen is rapidly purged from the CRDM housings due to leakage past the seals, crack growth occurs only during the first few weeks of operation following a refueling outage, followed by no growth for the remaining period of operation when oxygen concentrations are low. This equates to six oxygen ingress events (irrespective of time between events) for the crack to grow through wall.

NRC inspectors including technical experts from NRC Headquarters performed a follow-up inspection to determine if the assumptions made by the licensee were conservative and the planned actions bounded those conservative assumptions. The inspectors reviewed a variety of documents associated with crack growth and inspection intervals. The inspectors noted the following statements included in the RCR and vendor documents related to the determination of the appropriate crack growth rate:

- The laboratory conducting the failure analysis concluded, it could not be conclusively determined if the beach marks corresponded to refueling outages, (i.e., 18 month cycle) or shorter periods as occurred during outages over the past 24 months
- Palisades CRDM-21 leaked at weld No. 3 in 2001. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 failure. In calculating the crack growth rate of this crack, one contractor utilized an interval between beach marks, which is much shorter than refueling outages. The intervals used are consistent with plant thermal cycles in which oxygen may or may not have been admitted into the CRDMs.
- A CRDM housing at Ft Calhoun leaked at weld No. 5 in 1990. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 Palisades failure. In calculating the crack growth rate of this crack, Ft Calhoun stated that the beach marks were related to refueling cycles. Ft Calhoun also performed calculations indicating that the oxygen level at the location of the flaw did

not change with time (including in response to refueling outages) because the CRDM housing was not vented. Ft Calhoun's evaluation indicated that oxygen levels at the vicinity of the crack would have begun to decline through diffusion and convection had the intervals between outages been much longer than 18 months. This is interpreted to mean that the beach marks at Ft Calhoun are in response to pressure/thermal cycles.

- In at least one instance, Palisades needed to repair the seals on a reactor coolant pump at a time other than an outage. This necessitated draining some of the water from the reactor coolant system and venting (admitting oxygen into) the CRDM housing. This represented an additional oxygen ingress event not included when determination of time to cracking is based on refueling outages.
- In its inspection plan, Palisades stated that it will inspect all CRDM housings over the next four refueling outages, i.e., the interval between inspections is one refueling outage

Based on the above review, the inspectors noted that there were certain non conservative statements contained in the RCR and the inspection plan. These included:

- The crack growth rate based on refueling outages was understated. If oxygen ingress is related to beach marks, given the oxygen ingress event which occurred to repair reactor coolant pump seals, six beach marks would occur in a maximum of five refueling intervals rather than the six refueling intervals that were used to calculate the crack growth rate in the RCR.
- The crack growth rate based on heat up and cool down cycles is overstated. The value in the root cause is based on 11 months. While six shutdowns did occur at the plant in 11 months several of these events did not result in pressure/temperature changes of the reactor coolant system. The appropriate timeframe is 24 months rather than 11.
- The inspection plan contains a non conservative statement: "However, once the crack has been initiated it propagates over four to five operating cycles prior to going through wall." While this statement does reflect one of the proposed theories for crack growth, sufficient evidence to demonstrate reasonable assurance that this theory is correct, and thereby overcome the non-conservatism of this statement, was not provided.

Despite the existence of the non conservatisms stated above, the inspectors concluded:

- Sufficient evidence to conclusively determine the rate of crack growth does not exist.
- Crack growth based on pressure/temperature cycles is the most conservative of the potential crack growth mechanisms. In the absence of reasonable assurance of the correctness of less conservative mechanisms, through wall crack growth in two years must be utilized for regulatory purposes.
- The licensee has not formally committed to any of the crack growth mechanisms discussed.
- The licensee's inspection program includes inspection of all of the CRDM housings over the next four refueling outages. Approximately 25% of the housings will be inspected during each outage. The inspection of 25% of the CRDM housings each interval is sufficient to indicate that, in the event no indications are found during a given inspection, that the probability that flaws exist in other housings is extremely low. As such, it may be considered that the inspection of approximately 25% of the

CRDM housings every refueling outage bounds all the crack growth rate mechanisms considered.

Overall, some weaknesses did exist in the site's assessment, but none of these issues arose above the level of a minor performance deficiency for the evaluations completed. With the corrective actions in place to monitor the CRDMs, the inspectors considered this approach to inspection to be both acceptable and sufficient justification to close this URI.

4OA6 Management Meetings

.2 Interim Exit Meetings

An interim exit was conducted for:

- The results of the selected issue follow-up inspection, with Mr. T. Vitali, Site Vice President on April 18, 2013.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

B. Davis, Engineering Director
O. Gustafson, Licensing Manager
T. Foudy, Engineering Supervisor
B. Williams, Engineer
B. Dotson, Licensing

LIST OF ITEMS OPENED, CLOSED, DISCUSSED

Closed

05000255/2012012-01	URI	TS for PCS Pressure Boundary Leakage
05000255/2012012-02	URI	Potential Inadequate Degradation Evaluation of CRDM Housings
05000255/2012012-03	URI	Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality

Opened and Discussed

None.

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

- EN-LI-118, Root Cause Evaluation Process, Revision 18
- SOP-1B, Primary Coolant System – Cooldown, Revision 15

WI0PCS-M-06, NSSS Walkdown, Revision 3

- C-PAL-01-02186, Root Cause Evaluation, Primary Coolant System Pressure Boundary Leakage CRD-21 Upper Housing Assembly

-CAP029079, Primary Coolant System Pressure Boundary Leakage CRD-21 Upper Housing, June 21, 2001

- CR-PLP-2012-05623, Root Cause Evaluation Report, CRD-24 Upper Housing Leak, Revision 2
- CR-PLP-2013-01500, PCRS Condition Summary (NRC identified Criterion V violation), April 3, 2013

CR-PLP-2013-01134, PCRS Condition Summary, (Criterion XVI Violation), March 15, 2013

- PLP-RPT-13-00007, Laboratory Analysis of Leaking CRDM #24 Housing from Palisades, Revision 0
- PLP-RPT-12-00123, Examination of Cracks in CRDM Housing #24, Revision 0
- PLP-RPT-13-00009, Summary of Technical Documents Addressing the CRDM Housing 24 cracking at the Palisades Nuclear Plant, Revision 0
- PLP-RPT-13-00006, CRDM Housing at the Palisades Nuclear Plant – Recommended Future Actions, Revision 0
- PLP-RPT-12-0012, Evaluation of Residual Stresses in Flaw in CRD Housing Weld Overlay – Palisades Nuclear Plant, Revision 0
- PLP-RPT-12-00121, Evaluation of Thermal Stresses at Flaw Location in CRD Upper Housing – Palisades Nuclear plant, Revision 0
- PLP-RPT-12-00128, Prior Evaluations of Palisades CRDM Housing, Revision 0
- PLP-RPT-12-00125, Leakage Calculation for CRDM Housing, Revision 0
- PLP-RPT-12-00124
- LPI Report A12315-LR-003, Evaluation of Inside Surface Stresses above Sub-surface Flaws at Flaw Location in CRDM #24 Upper Housing – Palisades Nuclear plant, Revision 0
- EA-EAR-2001-0373-04, Owner's Review of SI "Evaluation of Leakage from Circumferential and Axial Through-wall Cracks in Lower CRDM Housing", July 22, 2001
- EA-EAR-2001-0426-01, CRD Upper Housing Redesign, January 17, 2002
- EA-C-PAL-01-2186-02, CRD Upper Housing and Nozzle Weld Susceptibility Comparison, Revision 1
- ANP-2547NP, Transgranular Stress Corrosion Cracking of Austenitic Stainless Steels in CRDM Applications, Revision 1
- Project RP-1063, Supplier Verification Deficiency Reports, December 2001/January2002
- WPS 1149-3, Welding Procedure Specification (GTAW), Revision 3
- WCAP-16000, Review of the Root Cause Evaluation for Leakage from Palisades CRD-21 Upper Housing Assembly C-PAL-01-2186, October 2003

LIST OF ACRONYMS USED

CRDM	Control Rod Drive Mechanism
ID	Inside Diameter
GPM	Gallons per Minute
LOCA	Loss of Coolant Accident
NCV	Non-Cited Violation
NDE	Non-Destructive Examination
OD	Outer Diameter
PCS	Primary Coolant System
RCR	Root Cause Report
RCT	Root Cause Team
SCAQ	Significant Condition Adverse to Quality
SDP	Significance Determination Process
SIT	Special Inspection Team
TGSCC	Transgranular Stress Corrosion Cracking
TS	Technical Specification
URI	Unresolved Item
UT	Ultrasonic Examination

Hills, David

From: Sellers, Loretta
Sent: Tuesday, May 07, 2013 11:19 AM
To: Giessner, John; Lennartz, Jay; Shah, Swetha; Betancourt, Diana; Taylor, Thomas; Scarbeary, April; Hernandez, Cammie; Alley, David; Lupold, Timothy
Cc: Hills, David; Sanchez Santiago, Elba; Holmberg, Mel; Lara, Julio
Subject: Palisades Input to DRP Report 2013 002
Attachments: Palisades Input to DRP Report 2013 002 URI EMS 5-6msh.docx

Importance: High

The subject document has been completed and has been submitted to DPC to be declared in ADAMS. A Word version of the document has been attached and the document has been assigned Accession #ML13127A220.

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U.S.NRC
Protecting People and the Environment

C-42

UNITED STATES
NUCLEAR REGULATORY COMMISSION
LISLE, IL 60532-4352

May 7, 2013

MEMORANDUM TO: Thomas Taylor
Senior Resident Inspector
Palisades Nuclear Plant

FROM: David Hills, Chief /RA by M. Holmberg for/
Engineering Branch 3
Division of Reactor Safety

SUBJECT: PALISADES NUCLEAR PLANT
DRS INPUT TO INTEGRATED REPORT 05000255/2013002

Enclosed is the report input for the Palisades Nuclear Plant, Inspection Report 05000255/2013002. This report input documents completion of our review of Unresolved Items 05000255/2012012-01, "TS for PCS Pressure Boundary Leakage," 05000255/2012012-02, "Potential Inadequate Degradation Evaluation of CRDM Housings," and 05000255/2012012-03, "Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality." This report also completes one sample of the Problem Identification and Resolution, Selected Issue Follow-up inspection in accordance with IP 71152. I have reviewed this input to confirm compliance with Inspection Manual Chapter (IMC) 0612 and IMC 0305. This input is ready for inclusion into the integrated report and dissemination to the public.

Please input the following post Inspection Data into RPS:

Inspection Procedure	**Procedure Status** – see below: Incomplete, Complete, Complete by reference, Complete-full sample not available, Complete – opportunity to apply procedure not available, Not Applicable.	**Sample Size** – As documented in Scope Section If less than full sample size documented in the report input, the inspector must provide a justification below to enter into RPS and support the procedure status selected
71152	Complete	One Sample

	Cornerstone (IE, MS, BI, EP, OR, PR, MISC)	**Cross utting Aspect**(H.n(i), P.n(i),S.n(i))	**Responsible Person/Owner**	**Procedure or TI** (71111.07T)	**RPS Branch Code (e.g. closeout responsibility)** EB1 3820 EB2 3870 EB3 3840 PST (RP) 3860 PSB (SG) 3850 OB 3810
NCV-XXX	IE	n/a	E. S. Santiago	71152	3820
NCV-XXX	IE	H.1(b)	E. S. Santiago	71152	3820

Enclosure: Input to Inspection Report 05000255/2013002
cc w/encl: J. Giessner, Chief
C. Hernandez, Site Admin Assistant
CONTACT: E. Sanchez Santiago, DRS
(630) 829-9715

DOCUMENT NAME: Palisades Input to DRP Report 2013 002 URI EMS 5-6msh.docx
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NAME	MHolmberg for ESanchezSantiago:ls		DAlley via email		M Holmberg for DHills		TLupold via email	
DATE	5/07/13		4/26/13		5/06/13		4/26/13	

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Cover Letter

_X_ Green findings involving a violation were identified. Include the following:

Based on the results of this inspection, two NRC-identified findings of very low safety significance (Green) were identified. These findings were determined to involve a violation of NRC requirements. However, because of the very low safety significance and because the issues were entered into your Corrective Action Program, the NRC is treating the issues as Non-Cited Violations, in accordance with Section 2.3.2 of the NRC Enforcement Policy.

TITLE PAGE

Inspectors: D. Alley, Senior Materials Engineer
E. Sanchez Santiago, Reactor Inspector

SUMMARY OF FINDINGS

A. <u>NRC-Identified and Self-Revealed Findings</u>

Cornerstone: Initiating Events

- <u>Green</u>. A self-revealing (Green) finding with associated Non-Cited Violations (NCVs) of 10 CFR Part 50, Appendix B, Criterion XVI, and Technical Specification (TS) 3.4.13, Primary Coolant System (PCS) Operational Leakage, was identified for failure to take corrective actions to prevent recurrence of Control Rod Drive Mechanism (CRDM) cracking and leakage, a significant condition adverse to quality (SCAQ), and resulting in operation of the reactor with PCS pressure boundary leakage. Specifically, for Criterion XVI the licensee failed to include the internal CRDM housing weld build-up area within the scope of corrective actions taken for a 2001 CRDM through wall leak on CRDM-21, caused by transgranular stress corrosion cracking (TGSCC). Subsequently, a through wall leak recurred in the weld build-up area on CRDM-24 in 2012 due to TGSCC. As a result, the licensee operated with PCS pressure boundary leakage, which is not allowed by TS 3.4.13. Further, because the licensee was not aware that the leakage was PCS pressure boundary leakage, the licensee did not implement the associated TS action statement. The licensee replaced CRDM-24 upper housing and wrote CR-PLP-2013-01134. Additional corrective actions are described in NRC Inspection Report 05000255/2012012.

  The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone objective to limit the likelihood of events that upset plant stability. The issue was associated with the attribute of equipment performance. Specifically, the licensee did not take adequate corrective actions to prevent recurrence of leakage in CRDM housings, which represents pressure boundary leakage. In accordance with Table 2, "Cornerstones Affected by Degraded Condition or Programmatic Weakness," of IMC 609, Attachment 4, "Initial Characterization of Findings," issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a

Primary System Loss of Coolant Accident (LOCA) initiator contributor. The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1, "Initiating Events Screening Questions," in IMC 0609, Attachment A, "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the slow rate of change for leakage for this cracking mechanism and this type of material. Type 316 stainless steel material under TGSCC will experience leakage rates well below a small break LOCA, which would be observed through the crack, alerting operators to take action to shut down the plant prior to experiencing a component rupture. The cause of this finding, non-conservative decision making, occurred over 10 years ago and is well outside of the nominal three year period in IMC 0612; and would not be indicative of current performance, because no other opportunities to identify the issue occurred during the previous three-year period. However more recently, the licensee exhibited non-conservative decision making with respect to addressing the potential for CRDM housing cracking and leakage during the recent root cause (Section 4OA2.3 (b.2) of this report), resulting in another finding. This cross-cutting aspect will be captured through the other finding. (Section 4OA2.3(b.1))

- Green. The inspectors identified a Finding with an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, for the licensee's failure to accomplish quality activities in accordance with the prescribed procedures. Specifically, the licensee failed to adequately evaluate and document the generic implications of the cause of the 2012 cracking identified in control rod drive mechanism (CRDM)-24 in accordance with Procedure, EN-LI-118, "Root Cause Evaluation." This issue was entered into the licensee's Corrective Action Program under CR-PLP-2013-01500. Subsequently, the licensee decided to revise the inspection plan to add additional corrective actions to inspect a sample of welds No. 3 and No. 4 for transgranular stress corrosion cracking (TGSCC) during the upcoming refueling outage.

  The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because the inspectors answered "yes" to the More-than-Minor screening question, "if left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern"? Specifically, absent NRC identification, the licensee would not have completed further evaluations or inspections of CRDM housing welds, which could have resulted in additional CRDM housing failure and leakage by TGSCC. In accordance with Table 2, "Cornerstones Affected by Degraded Condition or Programmatic Weakness," of IMC 609, Attachment 4, "Initial Characterization of Findings," issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System Loss of Coolant Accident (LOCA) initiator contributor. The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1, "Initiating Events Screening Questions," in IMC 0609, Attachment A, "The Significance Determination Process (SDP) for Findings At-Power," issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the slow rate of change for leakage for this cracking mechanism and this type of material. Type 316 stainless steel material under TGSCC will experience leakage rates well below a small break LOCA, which would be observed through the crack, alerting operators to take action to shut down the plant prior to experiencing a component rupture. The inspectors determined that the primary cause of the failure to adequately consider welds No. 3 and No. 4 in the generic implications section of the root cause report (RCR) related to the cross-cutting component of Human Performance, Decision Making, because licensee staff did not use conservative assumptions in decision making. Specifically, the licensee did not use conservative assumptions when excluding welds No. 3 and No. 4 as being susceptible to TGSCC when there was not enough information to exclude them from consideration. (Item H.1(b)). (Section 4OA2.3(b.2))

B. Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAILS

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (71152)

.3 Selected Issue Follow-up Inspection: Through Wall Leakage of CRDM-24 (This inspection is part of the additional inspections referenced in the Palisades Deviation letter.)

a. Inspection Scope

On August 12, 2012, the licensee shut down the plant to investigate an increase in unidentified leakage. The source of the leakage was determined to be a crack in CRDM-24. The NRC dispatched a special inspection team (SIT) to review the CRDM-24 leakage event. The results of that inspection were provided in Inspection Report 05000255/2012012. The licensee completed an evaluation to determine the cause of the cracking (CR-PLP-2012-05623).

From March 4, 2013, to March 15, 2013, the inspectors completed one inspection sample regarding problem identification and resolution based upon review of the licensee's RCR contained in corrective action document CR-PLP-2012-05623.
In addition, the inspectors performed reviews related to three Unresolved Items (URIs) identified during the SIT inspection:

- URI 05000255/2012012-01; Technical Specification (TS) for PCS Pressure Boundary Leakage. (The closure of this URI is documented in Section 4OA2.3 (b.1) of this report.);
- URI 05000255/2012012-02; Potential Inadequate Degradation Evaluation of CRDM Housings. (The closure of this URI is documented in Section 4OA5.1 of this report.); and
- URI 05000255/2012012-03; Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality. (The closure of this URI is documented in Section 4OA2.3 (b.1) of this report.).

The inspectors reviewed the licensee's actions in accordance with performance attributes identified in IP 71152. Specifically, the inspectors reviewed licensee corrective action records to determine whether: (1) the problems were accurately identified; (2) operability and reportability were adequately ascertained; (3) extent of condition and generic implications were appropriately addressed; (4) classification and prioritization of the problem were commensurate with safety significance; (5) root and contributing causes were identified; (6) corrective actions were appropriately focused to correct the problem; and (7) timely corrective actions were completed or proposed commensurate with the safety significance of the issues.

b. Findings

.1 Failure to Take Corrective Actions to Prevent Recurrence of CRDM Housing Cracking and Leakage

Introduction: A self-revealing Green Finding with associated NCV of 10 CFR Part 50, Appendix B, Criterion XVI, and TS 3.4.13 PCS Operational Leakage, was identified for failure to take corrective actions to prevent recurrence of CRDM cracking and leakage, a SCAQ, and resulting in operation of the reactor with PCS pressure boundary leakage. Specifically, for Criterion XVI, the licensee failed to include the internal CRDM housing weld build-up area within the scope of corrective actions taken for a 2001 CRDM through wall leak on CRDM-21 caused by TGSCC. Subsequently, a through wall leak recurred in the weld build-up area on CRDM-24 in 2012 due to TGSCC. As a result, the licensee operated with PCS pressure boundary leakage, which is not allowed by TS 3.4.13. Further, because the licensee was not aware that the leakage was PCS pressure boundary leakage, the licensee did not implement the associated TS action statement.

Description: In 2001, the licensee discovered a steam leak in the housing of CRDM-21 caused by a through-wall TGSCC at CRDM housing weld No. 3, which was located just below the weld build-up region (weld No. 5). Weld No. 5 consists of a weld material deposit applied to the inside diameter (ID) of the CRDM housing which provides for alignment of the CRDM. This issue was categorized as a SCAQ by the licensee (CPAL0102186) because it represented a break in the reactor system pressure boundary. The licensee's root cause evaluation was documented in RCR/C-PAL-01-02186 and concluded that the cracks in CRDM-21 were caused by TGSCC, which occurred in areas of heavy grinding or machining tool marks. Specifically, this leak was the result of an ID initiated, axially oriented, transgranular crack in the austenitic stainless steel housing material. The failure analysis performed in response to this event identified both axial and circumferential cracks associated with weld No. 3. Extent of condition inspections revealed additional, non-through wall cracks associated with weld No. 3 in 41 of the 44 remaining housings for a total of 42 of 45 housings containing cracks.

In response to the 2001 cracking, Palisades replaced all 45 CRDM housings with housings thought to be more resistant to cracking. Principle changes included:

- Elimination of weld No. 2;
- Relocation of weld No. 3 to a higher location thereby minimizing the deposition of crud in the gap between the weld and the bottom plate of the rack and pinion assembly;
- Reduction in residual stresses and cold work on welds by requiring better surface finishes, and
- Use of heat sink welding to reduce ID residual tensile stresses.

In January of 2002, an NRC SIT (reference IR 50-2555/01-15) reviewed the licensee proposed corrective actions associated with the through-wall leakage of the CRDM-21 housing caused by TGSCC. The 2001 RCR reviewed by the NRC stated the action to prevent recurrence was to "develop and implement an inspection plan to address areas and components identified in Attachment C-Extent of Condition." One of the components included in Attachment C was the CRDM. The recommended action was to perform volumetric inspection of the welds contained in the CRDM. Subsequently, the licensee decided to change this action and exclude weld No. 5.

Following the subsequent 2012 CRDM-24 leak, the licensee determined the leak occurred because of a through-wall crack adjacent to weld No. 5. The licensee formed a root cause team (RCT) staffed with licensee personnel and augmented with input from vendors. The root cause investigation was conducted in accordance with site procedure EN-LI-118, "Root Cause Evaluation Process" and was documented in root cause analysis report CR-PLP-2012-05623. In this report, the licensee's RCT determined that the probable cause of the cracking was:

> "Stresses in the weld build up area due to manufacturing irregularities and misalignments between CRDM-24 upper housing, support tube, and the associated reactor head penetration/CRDM nozzle. Based on lack of cracking found in the other eight upper housings tested, the failed CRDM-24 upper housing contains an as-yet unidentified additional stress."

The RCT also identified the following contributing cause:

> "TGSCC initiating within the internal weld build-up material of CRDM-24. The through wall crack initiated in the weld material and then propagated through the base metal until a leak developed in the outer diameter (OD) witness band region at the base of the ID weld build up."

This conclusion was based upon destructive and non destructive examinations (NDE) completed on a section of the failed housing, which included the through-wall flaw. The RCT also relied upon vendor technical reports assessing the results of the NDE as well as vendor calculations related to the stresses in the CRDM housings.

To determine the extent of condition, the licensee performed ultrasonic (UT) examinations of weld No. 5 on eight additional CRDM housings. The licensee selected these housings based on being in a similar location on the head as CRDM-24, and previous cracking having been identified in some of these housings prior to the replacement of the CRDM upper housings and seal housings in 2002. The inspectors concluded that this was an adequate sample for an initial extent of condition review based upon the concept that, in light of eight negative exams, the statistical probability of a flaw in the remaining CRDM housings was very low. Additionally, the licensee planned to conduct examinations of more housings during the next refueling outage.

The inspectors concluded that the licensee actions following the 2001 leak were not adequate because the appropriate actions to preclude recurrence were within the licensee's ability to foresee and implement. Specifically, the inspectors concluded that the licensee did not effectively implement corrective actions for the 2001 CRDM housing leak resulting in the 2012 CRDM-24 housing leak.

Licensee corrective actions taken in response to the 2001 event were limited to butt welds. The inspectors reviewed the licensee actions to determine if they had been sufficient to eliminate one of the three necessary factors to cause TGSCC on the CRDM housings: (1) a susceptible material, (2) a corrosive environment and (3) tensile stress. The inspectors identified that the licensee had failed to eliminate one or more of the necessary factors at weld No. 5 (which was not a butt weld) to preclude TGSCC in the replacement housing.

Specifically:

- The licensee's 2001 RCR documented that weld No. 5 is exposed to essentially the same environment as the weld that experienced the cracking (corrosive environment remained unchanged);

- No analysis was completed on the stress conditions for weld No. 5 prior to approving the modified replacement housing design (the potential for residual tensile weld stresses on ID of CRDM surface was not ruled out by analysis and therefore, should have been considered);

- Fabrication restrictions to prohibit grinding were not applied to weld No. 5 (grinding promotes residual tensile stress state on ID of CRDM surface);

- Machining was performed on weld No. 5 during the fabrication process in order to achieve the dimensions and geometry specified in the design. This process induced cold work stresses in the weld; and

- Material was changed from Type 347 to Type 316 stainless steel (both materials are essentially equally susceptible to TGSCC).

Also, in 1991, the Fort Calhoun plant had experienced through-wall leakage due to TGSCC at weld No. 5 of its CRDM housings (same housing design) and this operational experience had been reviewed by the licensee and dismissed. In the licensee's 2001 root cause evaluation, the licensee reviewed the weld build-up region failure by TGSCC at Fort Calhoun and concluded it would not occur at Palisades. This conclusion was based on the assumption that a higher oxygen environment (more aggressive environment) would exist in the Fort Calhoun housings than in the inservice Palisades housings. However the licensee did not confirm this assumption, nor did the licensee perform additional testing to determine if the environment of its inservice housings was sufficiently benign to prevent TGSCC. The licensee's 2012 RCT documented that due to organizational/ programmatic weakness at Palisades, the 1991 Fort Calhoun operating experience was not adequately utilized to include inspection of the weld No. 5. Similarly, the inspectors identified that the licensee had missed a key opportunity to implement effective corrective actions that could have prevented recurrence of the 2001 leakage event and had elected not to pursue that aspect further. Specifically, in EA-EAR-2001-0426-01, the licensee considered fabricating the replacement housings with Inconel 600 material because it was much more resistant to TGSCC, but ultimately decided not to do so. Additionally, various vendor reports were generated related to this issue in the mid 2000's. Those reports documented the potential susceptibility of weld No. 5 to TGSCC based upon a review of the CRDM housing conditions and available operating experience. The reports also noted that weld No. 5 was not inspected in any of the housings in 2001. One report in 2003 noted that weld No. 5 should have been examined as part of the action from the 2001 events since it was similar to Fort Calhoun. The issuance of these documents represented another opportunity for the licensee to identify the susceptibility of weld No. 5 to TGSCC prior to the cracking in CRDM-24.

The inspectors concluded the corrective actions taken in response to the 2001 CRDM through wall leak from TGSCC, a SCAQ, were not effective to preclude repetition. In particular, a through wall leak did recur on a CRDM from TGSCC. This issue was within the licensee's ability to foresee and correct; therefore, the issue was a performance deficiency. During the 2012 NRC special inspection, the NRC identified an URI for the TS pressure boundary leak. Technical Specifications LCO 3.4.13 does not allow any primary coolant system (PCS) pressure boundary leakage. In particular, TS Basis B3.4.13 "PCS Operational Leakage," explains that "No pressure boundary leakage from within the primary coolant pressure boundary is allowed, being indicative of material degradation. Leakage of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher leakage. Violation of this LCO could result in continued degradation of the primary coolant pressure boundary." Further, Action B, associated with this LCO, requires shutdown to Mode 3 in 6 hours and Mode 5 in 36 hours for such leakage. The licensee determined the CRDM-24 leakage commenced on or around July 14, 2012, and the plant continued to operate in this condition until August 12, 2012. Because the licensee was not aware of the existence of pressure boundary leakage, it failed to shut down the unit in six hours for a pressure boundary leak as required by TS 3.4.13 Action B. The NRC previously assessed the site's action for increasing unidentified leakage as part of the SIT. The NRC determined, at the time of higher unidentified leakage, the site took appropriate actions to attempt to locate the leak, eventually shutting down around .3 gallons per minute (gpm) leakage (earlier than the TS value of 1 gpm value for unidentified leakage). The licensee did not identify the source of the leakage as pressure boundary leakage until the shutdown on August 12, 2012, when a tour near the vessel head revealed the leaking housing. The pressure boundary leakage resulted in a TS violation and was due to the performance deficiency associated with the above mentioned Criterion XVI violation.

Based on the review discussed above, URIs 05000255/2012012-01 "TS for PCS Pressure Boundary Leakage" and 05000255/2012012-03 "Potential Failure to Take Corrective Actions to Prevent Recurrence of a Significant Condition Adverse to Quality" are closed.

Analysis: The inspectors determined that the licensee's failure to prevent recurrence of TGSCC of the CRDM housings (a SCAQ) that resulted in a violation of TS was a performance deficiency that warranted a significance evaluation. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone objective to limit the likelihood of events that upset plant stability. The issue was associated with the attribute of equipment performance. Specifically, the licensee did not take adequate corrective actions to prevent recurrence of leakage in CRDM housings, which represents pressure boundary leakage. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness," of IMC 609, Attachment 4, "Initial Characterization of Findings," issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609, Attachment A, "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the slow rate of change for leakage for this cracking mechanism and this type of material. Type 316 stainless steel material under TGSCC will experience leakage rates well below a small break LOCA, which would be observed through the crack, alerting operators to take action to shut down the plant prior to experiencing a component rupture.

The cause of this finding, non-conservative decision making, occurred over ten years ago and is well outside of the nominal three year period in IMC 0612; and was not indicative of current performance, because no other opportunities to identify the issue occurred during the previous three year period. However more recently, the licensee exhibited non-conservative decision making with respect to addressing the potential for CRDM housing cracking and leakage during the recent root cause (Section 4OA2.3 (b.2) of this report), resulting in another finding. This cross-cutting aspect will be captured through the other finding.

Enforcement: During this inspection, the inspectors identified two NCVs of NRC requirements:

- Title 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that, for significant conditions adverse to quality, the cause of the condition is determined and corrective action taken to preclude repetition; and

Contrary to the above, as of August 12, 2012, the licensee had failed to take corrective actions to preclude repetition for a SCAQ. Specifically, on June 21, 2001, the licensee discovered a through wall leak in CRDM-21 due to TGSCC and failed to reasonably include weld No. 5 in the corrective actions, which resulted in a subsequent through wall leak in CRDM-24 due to TGSCC.

- TS LCO 3.4.13 requires PCS operational leakage be limited to "No pressure boundary LEAKAGE" when in Modes 1 through 4.

Contrary to the above, on or around July 14, 2012, PCS pressure boundary leakage at CRDM-24 existed while in Mode 1. Further, because the licensee was not aware that the leakage was PCS pressure boundary leakage, the licensee did not implement the associated TS action statement.

As a result of the second through wall leak, the licensee took corrective actions, which included the development of an inspection plan that would inspect weld No. 5 every outage until all CRDM housings were inspected.

Because these violations were of very low safety significance and were entered into the licensee's Corrective Action Program as CR-PLP-2013-01134, these violations are being treated as an NCVs, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000255/2013002-XX; Failure to Take Corrective Action to Prevent Recurrence of CRDM Pressure Boundary Leakage).

.2 Failure to Adequately Address the Generic Implications of the Cracking identified in CRDM-24

Introduction: The inspectors identified a Finding with an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, for the licensee's failure to accomplish quality activities in accordance with the prescribed procedures. Specifically, the licensee failed to adequately evaluate and document the generic implications of the cause of the cracking identified in CRDM-24 in accordance with Procedure EN-LI-118, "Root Cause Evaluation." This issue was entered into the licensee's corrective action program under CR-PLP-2013-05623.

Description: As a result of the cracking identified in CRDM-24, which was characterized as a SCAQ, the licensee performed a root cause evaluation in accordance with Procedure EN-LI-118. This procedure was identified as quality related and served to implement control pursuant to the licensee's Quality Assurance Program. While reviewing the 2012 RCR (CR-PLP-2013-05623) related to the cracking identified in CRDM-24, the inspectors identified that the licensee had not appropriately considered the generic implications of the cracking in the extent of condition review. The licensee's proposed corrective actions, as a result of the 2012 RCR, narrowly focused on weld No. 5, instead of also including broader actions to ensure other CRDM housing welds were fit for their intended service life. These corrective actions consist of performing inspections of weld No. 5 on all CRDM housings.

On March 13, 2013, the inspectors requested that the licensee provide the bases for excluding other CRDM housing welds (weld No. 3 below weld No. 5 and weld No. 4 above weld No. 5) from the 2012 RCR scope of planned corrective actions. On March 29, 2013, the licensee provided additional information to justify excluding these welds from the scope of the corrective actions. The licensee credited the corrective actions associated with the modifications to the CRDM housing design completed in 2001 as the basis to exclude housing welds No. 3 and No. 4 from additional actions to identify the extent of TGSCC. The corrective actions taken in 2001 included performing heat sink welding, which is a methodology used to reduce the stresses on the inner ID of the weld. The licensee also changed the design to reduce design stresses at weld No. 3 and specified a smoother surface finish (RMS 125) to reduce potential crack initiation points. The licensee stated that these actions would produce compressive stresses on the ID of welds No. 3 and No. 4 making them immune from cracking. The inspectors acknowledged that these actions would reduce the tensile stress at the ID surface and thus reduce the probability of initiating TGSCC. However, the information provided did not demonstrate that TGSCC would not occur because it did not demonstrate that tensile stress would be eliminated at the ID surface during operation.

The inspectors identified that the three factors required for TGSCC could still be present at welds No. 3 and No. 4 as follows:

- Corrosive environment – Weld No. 3 would operate in a similar environment as weld No. 5 of the CRDM housing. Weld No. 4 would be exposed to a lower operating temperature than weld No. 5, however, TGSCC can still occur at 250 degrees Fahrenheit as evidenced by the Palisades previous operating experience with cracking identified in the seal housings that operate at even lower temperatures;

- Susceptible material – Welds No. 3 and No. 4 are composed of the same weld filler and base metal materials as weld No. 5 (e.g., weld filler material consistent with the Type 316 stainless housing base metal). This material would be equally susceptible to TGSCC, as the Type 347 stainless steel and weld filler materials used in the pre-2001 CRDM housing design that developed a through wall leak caused by TGSCC at weld No.3; and

- Tensile stresses – While it is assumed that the corrective actions taken in response to the 2001 leak will reduce the potential for tensile stresses to exist on the inner surface of CRDM housings at welds No. 3 and No. 4, especially in light of repairs made to welds No. 3 and No. 4, it had not been conclusively demonstrated that these tensile stresses have been eliminated. As such, when evaluating welds No. 3 and No. 4 for applicability to the 2012 root cause, it was not reasonable to conclude that tensile stresses were not present; and therefore, the potential for TGSCC had been eliminated.

The 2012 RCR discussed manufacturing irregularities and misalignment between CRDM-24 and the support tube, seismic supports, and the associated reactor head penetration/CRDM nozzle as potential source of stresses leading to cracking. However, the RCR also stated that "based on the lack of cracking found in the other eight upper housings tested, the failed CRDM-24 upper housing contains an as-yet unidentified additional stress." Because the cause of the additional stress was not identified, the licensee had not established a basis in the RCR to exclude welds No. 3 and No. 4 from the extent of condition review (e.g., potential generic implications). In 2001, assumptions on crack growth rate and inspection intervals for welds No. 3 and No. 4 were made based on the information known at the time. The 2001 crack went through-wall after the CRDM was in service for 30 years and the cracking was widespread among the other CRDM housings. In 2012, the crack propagated through-wall after the CRDM was in service for 11 years and the cracking did not appear as widespread. Though TGSCC was a factor in both cracking events, there are still unknowns associated with the 2012 incident. The unknown additional stresses, as well as the time the CRDM was inservice before cracking in 2012, represent key differences as related to the cracking identified in 2001. In the 2012 RCR, the licensee did not consider these or other potential differences between the two incidents when determining not to include welds No. 3 and No. 4 in the evaluation and documentation of the generic implications of the root and contributing causes and therefore, did not provide a justification for excluding welds No. 3 and No.4 from this evaluation or corrective actions.

The inspectors identified that the licensee had not followed Procedure EN-LI-118, in the root cause review of the CRDM-24 leak as documented in report CR-PLP-2013-05623. Section 5.5 (12)e of EN-LI-118 required that the licensee "perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affect other SSCs." Additional details are provided in the procedure on how to conduct and document the evaluation. In this case, the inspectors identified that the licensee had not addressed or documented a basis in RCR CR-PLP-2013-05623 to exclude welds No. 3 and No. 4 from the generic factors discussed above that led to the 2012 leak in CRDM-24 (e.g., TGSCC at weld No. 5). The licensee entered this issue into the Corrective Action Program as CR-PLP-2013-01500. Subsequently, the licensee decided to revise the inspection plan to add additional corrective actions to inspect a sample of welds No. 3 and No. 4 for TGSCC during the upcoming refueling outage.

Analysis: The inspectors determined that the failure to adequately evaluate and document the generic implications of the cause of the cracking identified in CRDM-24 in accordance with the root cause procedure EN-LI-118 was a performance deficiency that warranted a significance evaluation. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone attribute of equipment performance. The inspectors also answered "yes" to the More-than-Minor screening question, "if left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern"? Specifically, absent NRC identification, the licensee would not have completed further evaluations or inspections of CRDM housing welds, which could have resulted in additional CRDM housing failure and leakage by TGSCC. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness," of IMC 609, Attachment 4, "Initial Characterization of Findings," issued June 19, 2012, the inspectors checked the box under the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1, "Initiating Events Screening Questions," in IMC 0609, Attachment A, "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g., flaw tolerance) of the Type 316 stainless steel material such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a large component rupture.

The inspectors determined that the primary cause of the failure to adequately consider welds No. 3 and No. 4 in the generic implications section of the RCR related to the cross-cutting component of Human Performance, Decision Making, because licensee staff did not use conservative assumptions in decision making. Specifically, the licensee did not use conservative assumptions when excluding welds No. 3 and No. 4 as being susceptible to TGSCC and therefore include them in the generic implications section of the RCR. (Item H.1(b) of IMC 310).

Enforcement: During the inspection, the inspectors identified one NCV of NRC requirements:

- Title 10 CFR Part 50, Appendix B, Criterion V "Instructions, Procedures and Drawings requires in part, activities affecting quality shall be prescribed by documented procedures and shall be accomplished in accordance with these procedures.

Procedure EN-LI-118, "Root Cause Evaluation Process," Revision 17, states:

- Section 5.5 (12)e perform an extent of cause evaluation by reviewing the individual Root and Contributing causes for generic implications to establish whether the causes can affects other SSCs, organizations or work processes. Use the Two-Step Process in accordance with Attachment 9.7.

- Attachment 9.7: Determine whether the occurrence/consequence (problem) is isolated, or whether it has broader (generic or common mode) implications. Achieve this by asking the following questions:

  a) Could this happen to equipment that is similar in function, design, or service condition?

  b) Could this happen to a group of components? (components of the same construction or materials that could be similarly affected by one condition)?

- Attachment 9.7: Document the results of the above considerations. Include the following items in the write up:

  a) Generic Implications. (Is this problem/cause limited to this component/equipment, or does it apply to others as well)?

  b) Existing broader (generic/common mode) considerations.

- Section 5.5(15)(10)c and f: Document proposed corrective actions and due dates to address valid generic implications. If no corrective action is recommended for a valid generic implication then document the basis for this conclusion and any risk or consequence identified, as a result of taking no action.

Contrary to the above, from February 24, 2013, through April 18, 2013, the licensee failed to accomplish activities affecting quality in accordance with procedure EN-LI-118, which was being implemented to correct a SCAQ. Specifically, the licensee failed to accomplish Section 5.5 (12)e by not fully evaluating and documenting the existing broader (generic/common mode) considerations, extent of condition/cause associated with TGSCC at CRDM housing welds No. 3 and No. 4, including considering the susceptibility of the welds to TGSCC and the need to perform subsequent inspections or evaluations.

Subsequently, the licensee decided to revise the inspection plan to add additional corrective actions to inspect a sample of welds No. 3 and No. 4 for TGSCC during the upcoming refueling outage.

Because of the very low safety significance and because the licensee entered this issue into their Corrective Action Program (CR-PLP-2013-01500), it is being treated as an NCV consistent with Section 2.3.2 of the Enforcement Policy (NCV 05000255/2013003-xx; Failure to Adequately Address the Generic Implications of the Cracking Identified in CRDM-24).

4OA5 Other Activities

.1 (Closed) Unresolved Item 05000255/2012012-02: Potential Inadequate Degradation Evaluation of CRDM Housings (This inspection is part of the additional inspections included in the Palisades Deviation letter)

During a Special Inspection performed in August 2012, NRC inspectors identified an issue, which could not be resolved without additional information (URI). This issue was associated with the rate of growth of the crack which created the through wall leak in CRDM-24, discovered on August 12, 2012. Identification of this crack growth rate is significant in determining appropriate intervals for future inspections to provide reasonable assurance that CRDM housing leakage will not recur.

Preliminary failure analysis data available at the time of the inspection indicated that the observed cracking was due to TGSCC. Cracking of this type is normally due to the presence of oxygen and chlorides at the location of the crack. When examining the fracture surface at the location the through-wall leak occurred, the licensee identified six concentric rings (beach marks) propagating in a radial direction from the ID out towards the OD of the housing. Beach marks are normally associated with fatigue failures and indicate the number of stress cycles from crack initiation to crack failure. In this case, there was no evidence that fatigue contributed to the failure. Despite the lack of evidence of fatigue, it was apparent that the crack, which resulted in the CRDM-24 leak, grew in increments. It was not, however, immediately apparent whether the increments were related to oxygen ingress (refueling outages) or temperature/pressure cycles (heatups/cooldowns).

At the time of the original inspection, five time intervals for through wall crack growth were under consideration. Two were based on literature crack growth data and three were based on interpretations of the beach marks. These time intervals were:

- Based on literature data, one contractor estimated that a 10% through wall flaw would require four years to reach 50% through wall.
- Based on literature data another contractor estimated the crack growth rate to be 2.1×10^{-5} in/hr or 0.18 in/yr. This is approximately three times faster than the crack growth rate proposed in the above mentioned rate.
- Based on the concept of oxygen ingress at refueling outages six cycles of 18 months duration would require nine years for the crack to grow through wall.
- Based on the concept of temperature/pressure cycles, the plant experienced six cold shutdowns in approximately two years preceding the crack. This equates to two years for the crack to grow through wall.
- Based on the concept that oxygen is required for crack growth and that oxygen is rapidly purged from the CRDM housings due to leakage past the seals, crack growth occurs only during the first few weeks of operation following a refueling outage, followed by no growth for the remaining period of operation when oxygen concentrations are low. This equates to six oxygen ingress events (irrespective of time between events) for the crack to grow through wall.

NRC inspectors including technical experts from NRC Headquarters performed a follow-up inspection to determine if the assumptions made by the licensee were conservative and the planned actions bounded those conservative assumptions. The inspectors reviewed a variety of documents associated with crack growth and inspection intervals. The inspectors noted the following statements included in the RCR and vendor documents related to the determination of the appropriate crack growth rate:

- The laboratory conducting the failure analysis concluded, it could not be conclusively determined if the beach marks corresponded to refueling outages, (i.e., 18 month cycle) or shorter periods as occurred during outages over the past 24 months.

- Palisades CRDM-21 leaked at weld No. 3 in 2001. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 failure. In calculating the crack growth rate of this crack, one contractor utilized an interval between beach marks, which is much shorter than refueling outages. The intervals used are consistent with plant thermal cycles in which oxygen may or may not have been admitted into the CRDMs.

- A CRDM housing at Fort Calhoun leaked at weld No. 5 in 1990. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 Palisades failure. In calculating the crack growth rate of this crack, Fort Calhoun stated that the beach marks were related to refueling cycles. Fort Calhoun also performed calculations indicating that the oxygen level at the location of the flaw did not change with time (including in response to refueling outages) because the CRDM housing was not vented. Fort Calhoun's evaluation indicated that oxygen levels at the vicinity of the crack would have begun to decline through diffusion and convection had the intervals between outages been much longer than 18 months. This is interpreted to mean that the beach marks at Fort Calhoun are in response to pressure/thermal cycles.

- In at least one instance, Palisades needed to repair the seals on a reactor coolant pump at a time other than an outage. This necessitated draining some of the water from the reactor coolant system and venting (admitting oxygen into) the CRDM housing. This represented an additional oxygen ingress event not included when determination of time to cracking is based on refueling outages.

- In its inspection plan, Palisades stated that it will inspect all CRDM housings over the next four refueling outages, i.e., the interval between inspections is one refueling outage.

Based on the above review, the inspectors noted that there were certain non conservative statements contained in the RCR and the inspection plan. These included:

- The crack growth rate based on refueling outages was understated. If oxygen ingress is related to beach marks, given the oxygen ingress event which occurred to repair reactor coolant pump seals, six beach marks would occur in a maximum of five refueling intervals rather than the six refueling intervals that were used to calculate the crack growth rate in the RCR.

- The crack growth rate based on heat up and cool down cycles is overstated. The value in the root cause is based on 11 months. While six shutdowns did occur at the plant in 11 months several of these events did not result in pressure/ temperature changes of the reactor coolant system. The appropriate timeframe is 24 months rather than 11.

- The inspection plan contains a non-conservative statement: "However, once the crack has been initiated it propagates over four to five operating cycles prior to going through wall." While this statement does reflect one of the proposed theories for crack growth, sufficient evidence to demonstrate reasonable assurance that this theory is correct, and thereby overcome the non-conservatism of this statement, was not provided.

Despite the existence of the non-conservatisms stated above, the inspectors concluded:

- Sufficient evidence to conclusively determine the rate of crack growth does not exist;
- Crack growth based on pressure/temperature cycles is the most conservative of the potential crack growth mechanisms. In the absence of reasonable assurance of the correctness of less conservative mechanisms, through wall crack growth in two years must be utilized for regulatory purposes;
- The licensee has not formally committed to any of the crack growth mechanisms discussed; and
- The licensee's inspection program includes inspection of all of the CRDM housings over the next four refueling outages. Approximately 25% of the housings will be inspected during each outage. The inspection of 25% of the CRDM housings each interval is sufficient to indicate that, in the event no indications are found during a given inspection, that the probability that flaws exist in other housings is extremely low. As such, it may be considered that the inspection of approximately 25% of the CRDM housings every refueling outage bounds all the crack growth rate mechanisms considered.

Overall, some weaknesses did exist in the site's assessment, but none of these issues arose above the level of a minor performance deficiency for the evaluations completed. With the corrective actions in place to monitor the CRDMs, the inspectors considered this approach to inspection to be both acceptable and sufficient justification to close this URI.

4OA6 Management Meetings

.2 Interim Exit Meetings

An interim exit was conducted for:

- The results of the selected issue follow-up inspection, with Mr. T. Vitale, Site Vice President on April 18, 2013.

The inspectors confirmed that none of the potential report input discussed was considered proprietary. Proprietary material received during the inspection was returned to the licensee or destroyed.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

B. Davis, Engineering Director
O. Gustafson, Licensing Manager
T. Foudy, Engineering Supervisor
B. Williams, Engineer
B. Dotson, Licensing

LIST OF ITEMS OPENED, CLOSED, DISCUSSED

Closed

05000255/2012012-01	URI	TS for PCS Pressure Boundary Leakage
05000255/2012012-02	URI	Potential Inadequate Degradation Evaluation of CRDM Housings
05000255/2012012-03	URI	Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality

Opened and Discussed

None

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

4OA2 Identification and Resolution of Problems (71152)

- EN-LI-118, Root Cause Evaluation Process, Revision 18

- SOP-1B, Primary Coolant System – Cooldown, Revision 15

- WI0PCS-M-06, NSSS Walkdown, Revision 3

- C-PAL-01-02186, Root Cause Evaluation, Primary Coolant System Pressure Boundary Leakage CRD-21 Upper Housing Assembly

- CAP029079, Primary Coolant System Pressure Boundary Leakage CRD-21 Upper Housing, June 21, 2001

- CR-PLP-2012-05623, Root Cause Evaluation Report, CRD-24 Upper Housing Leak, Revision 2

- CR-PLP-2013-01500, PCRS Condition Summary (NRC identified Criterion V violation), April 3, 2013

- CR-PLP-2013-01134, PCRS Condition Summary, (Criterion XVI Violation), March 15, 2013

- PLP-RPT-13-00007, Laboratory Analysis of Leaking CRDM #24 Housing from Palisades, Revision 0

- PLP-RPT-12-00123, Examination of Cracks in CRDM Housing #24, Revision 0

- PLP-RPT-13-00009, Summary of Technical Documents Addressing the CRDM Housing 24 cracking at the Palisades Nuclear Plant, Revision 0

- PLP-RPT-13-00006, CRDM Housing at the Palisades Nuclear Plant – Recommended Future Actions, Revision 0

- PLP-RPT-12-0012, Evaluation of Residual Stresses in Flaw in CRD Housing Weld Overlay – Palisades Nuclear Plant, Revision 0

- PLP-RPT-12-00121, Evaluation of Thermal Stresses at Flaw Location in CRD Upper Housing – Palisades Nuclear plant, Revision 0

- PLP-RPT-12-00128, Prior Evaluations of Palisades CRDM Housing, Revision 0

- PLP-RPT-12-00125, Leakage Calculation for CRDM Housing, Revision 0

- PLP-RPT-12-00124

- LPI Report A12315-LR-003, Evaluation of Inside Surface Stresses above Sub-surface Flaws at Flaw Location in CRDM #24 Upper Housing – Palisades Nuclear plant, Revision 0

- EA-EAR-2001-0373-04, Owner's Review of SI "Evaluation of Leakage from Circumferential and Axial Through-wall Cracks in Lower CRDM Housing," July 22, 2001

- EA-EAR-2001-0426-01, CRD Upper Housing Redesign, January 17, 2002

- EA-C-PAL-01-2186-02, CRD Upper Housing and Nozzle Weld Susceptibility Comparison, Revision 1

- ANP-2547NP, Transgranular Stress Corrosion Cracking of Austenitic Stainless Steels in CRDM Applications, Revision 1

- Project RP-1063, Supplier Verification Deficiency Reports, December 2001/January2002

- WPS 1149-3, Welding Procedure Specification (GTAW), Revision 3

- WCAP-16000, Review of the Root Cause Evaluation for Leakage from Palisades CRD-21 Upper Housing Assembly C-PAL-01-2186, October 2003

LIST OF ACRONYMS USED

CRDM	Control Rod Drive Mechanism
ID	Inside Diameter
GPM	Gallons per Minute
LOCA	Loss of Coolant Accident
NCV	Non-Cited Violation
NDE	Non-Destructive Examination
OD	Outer Diameter
PCS	Primary Coolant System
RCR	Root Cause Report
RCT	Root Cause Team
SCAQ	Significant Condition Adverse to Quality
SDP	Significance Determination Process
SIT	Special Inspection Team
TGSCC	Transgranular Stress Corrosion Cracking
TS	Technical Specification
URI	Unresolved Item
UT	Ultrasonic Examination

8

Craver, Patti

From:	Worosilo, Jannette
Sent:	Wednesday, July 11, 2012 8:03 AM
To:	Farnan, Michael
Subject:	FW: Harris SIT report IR 2012-008
Attachments:	HAR IR 12-008.docx

Michael,

Can you please review and send me your concurrence on the attached report?? The Branch Chief will like to sign it out today.

Thanks,

Jannette

From: Worosilo, Jannette
Sent: Tuesday, July 10, 2012 6:37 AM
To: Zeiler, John; Lessard, Patrick; Dodson, Jim; Steadham, Timothy; Farnan, Michael
Subject: Harris SIT report IR 2012-008

Please review and provide your concurrence on Harris IR 2012-008.

Thanks,

Jannette G. Worosilo
U.S. Nuclear Regulatory Commission
Region II - Atlanta, GA
Project Engineer
Division of Reactor Projects
Reactor Projects Branch 4
(404) 997-4485
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Releasable -

C43

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
245 PEACHTREE CENTER AVENUE NE, SUITE 1200
ATLANTA, GEORGIA 30303-1257

Mr. Christopher Burton, Vice President
Carolina Power and Light Company
Shearon Harris Nuclear Power Plant
P. O. Box 165, Mail Code: Zone 1
New Hill, North Carolina 27562-0165

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT - NRC SPECIAL INSPECTION REPORT 05000400/2012008

Dear Mr. Burton:

On May 30, 2012, the U.S. Nuclear Regulatory Commission (NRC) completed a reactive inspection pursuant to Inspection Procedure 93812, "Special Inspection" at your Shearon Harris reactor facility Unit 1. The enclosed inspection report documents the inspection results which were discussed on May 30, 2012, with you and other members of your staff.

The special inspection was commenced on May 7, 2012, in accordance with Management Directive 8.3, "NRC Incident Investigation Program," and Inspection Manual 0309, "Reactive Inspection Decision Basis for Reactors," based on the initial risk and deterministic criteria evaluation made by the NRC on April 24, 2012.

The special inspection reviewed the circumstances surrounding the failure of two safety-related main steam isolation valves (MSIVs) to close which occurred on April 21, 2012, and examined activities conducted under your license as they relate to safety and compliance with the Commission's rule and regulations and with the conditions of your license. The inspection started on May 7, 2012, and the preliminary inspection results were discussed with you and members of your staff on May 11, 2012. Subsequent onsite inspections were conducted May 16 – 18, 2012, to observe MSIV testing following your maintenance repairs, and further in-office reviews of post-maintenance testing results were conducted May 21 – 25, 2012.

No findings were identified during this inspection.

C L

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's Agencywide Document Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

Randall A. Musser, Chief
Reactor Projects Branch 4
Division of Reactor Projects

Docket Nos.: 50-400
License No.: NPF-63

Enclosure: NRC Inspection Report 05000400/2012008 w/Attachment: Supplemental Information

cc w/encl: (See page 3)

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's Agencywide Document Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

Randall A. Musser, Chief
Reactor Projects Branch 4
Division of Reactor Projects

Docket Nos.: 50-400
License No.: NPF-63

Enclosure: NRC Inspection Report 05000400/2012008 w/Attachment: Supplemental Information

cc w/encl: (See page 3)

X PUBLICLY AVAILABLE ☐ NON-PUBLICLY AVAILABLE ☐ SENSITIVE X NON-SENSITIVE
ADAMS: X ACCESSION NUMBER:____________________ X SUNSI REVIEW COMPLETE X FORM 665 ATTACHED

OFFICE	RII:DRP	RII:DRP	RII:DRP	RII:DRP	RII:DRP	RII:CCI	RII:HQ
SIGNATURE							
NAME	JDodson	RMusser	JZeiler	PLessard	JWorosilo	TSteadham	MFarnan
DATE	06/ /2012	06/ /2012	06/ /2012	06/ /2012	06/ /2012	06/ /2012	
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

OFFICIAL RECORD COPY DOCUMENT NAME: G:\DRPII\RPB4\HARRIS\REPORTS\2012 REPORTS\12-08\HAR IR 12-008.DOCX

cc w/encl:
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cc w/encl. (continued next page)

cc w/encl. (continued)
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9

Decision Documentation for Reactive Inspection (Deterministic-only Criteria Analyzed)		
PLANT: Duane Arnold Energy Center	EVENT DATE: 10/16/12	EVALUATION DATE: 10/17/12

Brief Description of the Significant Operational Event or Degraded Condition:
The plant is currently in a refueling and maintenance outage, which includes repair and recoating of the torus. It was during the performance of this activity that the work identified the need for additional tie off points for fall protection purposes. The new locations were discussed, without specificity, with radiation protection. This expanded scope of work was authorized but the areas were not surveyed by radiation protection. Ten workers became contaminated, with nine exhibiting an uptake of radioactive materials. The initial dose calculations indicate a maximum of 19 mrem to one of the workers.

REACTOR SAFETY

Y/N	**IIT Deterministic Criteria**
N/A	Led to a Site Area Emergency
	Remarks:
N/A	Exceeded a safety limit of the licensee's technical specifications
	Remarks:
N/A	Involved circumstances sufficiently complex, unique, or not well enough understood, or involved safeguards concerns, or involved characteristics the investigation of which would best serve the needs and interests of the Commission
	Remarks:
Y/N	**SI Deterministic Criteria**
N/A	Significant failure to implement the emergency preparedness program during an actual event, including the failure to classify, notify, or augment onsite personnel
	Remarks:
N/A	Involved significant deficiencies in operational performance which resulted in degrading, challenging, or disabling a safety system function or resulted in placing the plant in an unanalyzed condition for which available risk assessment methods do not provide an adequate or reasonable estimate of risk.
	Remarks:

RGIII

ML12300A310 Non Public

RADIATION SAFETY	
Y/N	**IIT Deterministic Criteria**
N	Led to a significant radiological release (levels of radiation or concentrations of radioactive material in excess of 10 times any applicable limit in the license or 10 times the concentrations specified in 10 CFR Part 20, Appendix B, Table 2, when averaged over a year) of byproduct, source, or special nuclear material to unrestricted areas
	Remarks: This event occurred inside the torus and did not constitute a radiological release to unrestricted areas.
N	Led to a significant occupational exposure or significant exposure to a member of the public. In both cases, "significant" is defined as five times the applicable regulatory limit (except for shallow-dose equivalent to the skin or extremities from discrete radioactive particles)
	Remarks: This event did not lead to a significant occupational exposure as the highest dose was 19 mrem (CEDE) to an occupational radiation worker. Furthermore, there was no exposure to members of the public.
N	Involved the deliberate misuse of byproduct, source, or special nuclear material from its intended or authorized use, which resulted in the exposure of a significant number of individuals
	Remarks: This event was caused by work in an area that was not surveyed by radiation protection and did not involve the misuse of radioactive material.
N	Involved byproduct, source, or special nuclear material, which may have resulted in a fatality
	Remarks: Affected workers were evaluated after the event and the event did not involve a fatality.
N	Involved circumstances sufficiently complex, unique, or not well enough understood, or involved safeguards concerns, or involved characteristics the investigation of which would best serve the needs and interests of the Commission
	Remarks: Radiological surveys were performed after the event and the radiological conditions are understood.

Y/N	AIT Deterministic Criteria
N	Led to a radiological release of byproduct, source, or special nuclear material to unrestricted areas that resulted in occupational exposure or exposure to a member of the public in excess of the applicable regulatory limit (except for shallow-dose equivalent to the skin or extremities from discrete radioactive particles)
	Remarks: This event occurred inside the torus and did not constitute a radiological release to unrestricted areas.
N	Involved the deliberate misuse of byproduct, source, or special nuclear material from its intended or authorized use and had the potential to cause an exposure of greater than 5 rem to an individual or 500 mrem to an embryo or fetus
	Remarks: This event was caused by work in an area that was not surveyed by radiation protection and did not involve the misuse of radioactive material.
N	Involved the failure of radioactive material packaging that resulted in external radiation levels exceeding 10 rads/hr or contamination of the packaging exceeding 1000 times the applicable limits specified in 10 CFR 71.87
	Remarks: This event did not involve packaging of radioactive material.
N	Involved the failure of the dam for mill tailings with substantial release of tailings material and solution off site
	Remarks: This event did not involve mill tailings.

Y/N	SI Deterministic Criteria
N	May have led to an exposure in excess of the applicable regulatory limits, other than via the radiological release of byproduct, source, or special nuclear material to the unrestricted area; specifically • occupational exposure in excess of the regulatory limits in 10 CFR 20.1201 • exposure to an embryo/fetus in excess of the regulatory limits in 10 CFR 20.1208 • exposure to a member of the public in excess of the regulatory limits in 10 CFR 20.1301
	Remarks: The maximum dose to any worker was 19 mrem (CEDE) and does not constitute an overexposure. Furthermore, this event did not involve declared pregnant workers or members of the public.
N	May have led to an unplanned occupational exposure in excess of 40 percent of the applicable regulatory limit (excluding shallow-dose equivalent to the skin or extremities from discrete radioactive particles)
	Remarks: The maximum dose for this event was <100 mR (SDE) or 500 times less than the regulatory limit.

N	Led to unplanned changes in restricted area dose rates in excess of 20 rem per hour in an area where personnel were present or which is accessible to personnel
	Remarks: General area dose rates in the work area were 5 mrem/hour and did not change during the event.
N	Led to unplanned changes in restricted area airborne radioactivity levels in excess of 500 DAC in an area where personnel were present or which is accessible to personnel and where the airborne radioactivity level was not promptly recognized and/or appropriate actions were not taken in a timely manner
	Remarks: Air samples collected reported that airborne radioactivity levels did not exceed 0.3 DAC in the area where the workers were present.
N	Led to an uncontrolled, unplanned, or abnormal release of radioactive material to the unrestricted area • for which the extent of the offsite contamination is unknown; or, • that may have resulted in a dose to a member of the public from loss of radioactive material control in excess of 25 mrem (10 CFR 20.1301(e)); or, • that may have resulted in an exposure to a member of the public from effluents in excess of the ALARA guidelines contained in Appendix I to 10 CFR Part 50
	Remarks: This event occurred inside the torus and did not constitute a radiological release to unrestricted areas.
N	Led to a large (typically greater than 100,000 gallons), unplanned release of radioactive liquid inside the restricted area that has the potential for ground-water, or offsite, contamination
	Remarks: This event occurred inside the torus and did not constitute a radiological release to unrestricted areas.
N	Involved the failure of radioactive material packaging that resulted in external radiation levels exceeding 5 times the accessible area dose rate limits specified in 10 CFR Part 71, or 50 times the contamination limits specified in 49 CFR Part 173
	Remarks: This event did not involve packaging of radioactive material.
N	Involved an emergency or non-emergency event or situation, related to the health and safety of the public or on-site personnel or protection of the environment, for which a 10 CFR 50.72 report has been submitted that is expected to cause significant, heightened public or government concern
	Remarks: This event did not report or plan to report the event per 10 CFR 50.72.

SAFEGUARDS/SECURITY	
Y/N	**IIT Deterministic Criteria**
N/A	Involved circumstances sufficiently complex, unique, or not well enough understood, or involved safeguards concerns, or involved characteristics the investigation of which would best serve the needs and interests of the Commission
	Remarks:
N/A	Failure of licensee significant safety equipment or adverse impact on licensee operations as a result of a safeguards initiated event (e.g., tampering).
	Remarks:
N/A	Actual intrusion into the protected area.
	Remarks:
Y/N	**AIT Deterministic Criteria**
N/A	Involved a significant infraction or repeated instances of safeguards infractions that demonstrate the ineffectiveness of facility security provisions
	Remarks:
N/A	Involved repeated instances of inadequate nuclear material control and accounting provisions to protect against theft or diversions of nuclear material
	Remarks:
N/A	Confirmed tampering event involving significant safety or security equipment
	Remarks:
	Substantial failure in the licensee's intrusion detection or package/personnel search procedures which results in a significant vulnerability or compromise of plant safety or security
	Remarks:
Y/N	**SI Deterministic Criteria**
N/A	Involved inadequate nuclear material control and accounting provisions to protect against theft or diversion, as evidenced by inability to locate an item containing special nuclear material (such as an irradiated rod, rod piece, pellet, or instrument)
	Remarks:

N/A	Involved a significant safeguards infraction that demonstrates the ineffectiveness of facility security provisions
	Remarks:
N/A	Confirmation of lost or stolen weapon
	Remarks:
N/A	Unauthorized, actual non-accidental discharge of a weapon within the protected area
	Remarks:
N/A	Substantial failure of the intrusion detection system (not weather related)
	Remarks:
N/A	Failure to the licensee's package/personnel search procedures which results in contraband or an unauthorized individual being introduced into the protected area
	Remarks:
N/A	Potential tampering of vandalism event involving significant safety or security equipment where questions remain regarding licensee performance/response or a need exists to independently assess the licensee's conclusion that tampering or vandalism was not a factor in the condition(s) identified
	Remarks:

RESPONSE DECISION
USING THE ABOVE INFORMATION AND OTHER KEY ELEMENTS OF CONSIDERATION AS APPROPRIATE, DOCUMENT THE RESPONSE DECISION TO THE EVENT OR CONDITION, AND THE BASIS FOR THAT DECISION
DECISION AND DETAILS OF THE BASIS FOR THE DECISION: A reactive inspection is not warranted for this event. The event is currently being inspected by two health physicists from Region III that were on-site conducting baseline inspection procedures for the refueling outage.

BRANCH CHIEF REVIEW: ***/RA/ B. Dickson***	DATE: 10/18/12
TSS TEAM LEADER REVIEW: **/RA/ J. Lara**	DATE: 10/18/12
DIVISION DIRECTOR REVIEW: ***/RA/ S. West***	DATE: 10/22/12
DIVISION DIRECTOR REVIEW: ***/RA/ By K. O'Brien Acting For S. Reynolds/***	DATE: 10/25/12
ADAMS ACCESSION NUMBER ML12300A310 EVENT NOTIFICATION REPORT NUMBER (as applicable):	

DISTRIBUTION:		**Region**
Darrell Roberts	DRP Division Director	I
James Clifford	DRP Deputy Director	I
Chris Miller	DRS Division Director	I
Peter Wilson	DRS Deputy Director	I
Rick Croteau	DRP Division Director	II
William Jones	DRP Deputy Director	II
Terrence Reis	DRS Division Director	II
Harold Christensen	DRS Deputy Director	II
Steven West	DRP Division Director	III
Gary Shear	DRP Deputy Director	III
Steven Reynolds	DRS Division Director	III
Kenneth O'Brien	DRS Deputy Director	III
Kriss Kennedy	DRP Division Director	IV
Allen Howe	DRP Deputy Director (Acting)	IV
Thomas Blount	DRS Division Director	IV
Jeffrey Clark	DRS Deputy Director (Acting)	IV
Julio Lara	Branch TSS Team Leader	III
Doris Chyu	Reactor Engineer	III
Nicholas Valos	Senior Reactor Analyst	III
Laura Kozak	Senior Reactor Analyst	III
Dave Passehl	Senior Reactor Analyst	III
NRR_Reactive Inspection@nrc.gov		

From:	Miller, Geoffrey
To:	Vegel, Anton; Blount, Tom
Cc:	Kennedy, Kriss; Pruett, Troy; Clark, Jeff; Collins, Elmo; Graves, Samuel; Gepford, Heather; Deese, Rick; Howell, Art; Kirkland, John; Wingebach, Jacob
Subject:	Fort Calhoun Preliminary Red Response
Date:	Monday, March 19, 2012 3:25:00 PM

Tony/Tom,

I spoke with Corey Cameron at Fort Calhoun (acting for Susan Baughn this week) about Fort Calhoun's requested 14-day extension for their written response to the Preliminary Red finding. I told him the extension was granted, and I requested that the station document in their 10-day written response letter (requested by the Choice Letter) that they had declined a Regulatory Conference and would provide a written response by April 25, noting that they had requested and received a 14-day extension via telecom with me on March 19. I explained that by doing so, the revised due date would be appropriately reflected on the public docket. He said he understood and would ensure their letter contained this information. I also emphasized that a written response to the finding would not provide the same opportunity for clarifying questions and back-and-forth information exchange that a reg conference would, and so would not be the preferred mechanism for disputing a violation or its significance from an efficiency standpoint (though allowed per the Choice Letter). He said he understood this as well. Please let me know if you have questions or would like additional information.

Thank you,

Geoff

Release

C/46

RELEASE IN ENTIRETY

From: Kellar, Ray
To: Graves, Samuel; Miller, Geoffrey; Gepford, Heather; Maier, Christi; Clark, Jeff
Cc: Kennedy, Kriss; Pruett, Troy; Blount, Tom; Loveless, David
Subject: MC 0609 Preliminary Significance
Date: Tuesday, January 31, 2012 1:57:29 PM

FYI, Follow-up on Guidance for preliminary significance from MC 0609.01:

Four classifications:

White, Yellow, Red, or greater than Green

02.04.c Preliminary SERP Reviews - Greater Than Green Findings.

1. The "greater than Green" option is not expected to be the norm when characterizing the preliminary significance of findings.

2. The staff should make realistic assumptions in the bases for its significance determinations and should make a reasonable effort to determine a specific preliminary color in a timely manner. Every effort should be made during the peer review to resolve all differences and concerns.

3. The preliminary significance of a finding should be characterized as "potentially greater than Green" if the staff:

> (a) Is unable to determine a specific preliminary color because of the proximity to a color threshold, or
>
> (b) Lacks information to make reasonable assumptions, and the assumptions are influential to the preliminary significance result (i.e., will cause the color to vary).

When this option is used, the SDP basis provided to the licensee must be particularly clear and complete to identify where the staff lacks information to reach a final determination.

Ray L. Kellar, P.E.
Senior Enforcement Specialist
817-200-1121 work
817-200-1122 fax
Ray.Kellar@nrc.gov

Release in entirety

From: Loveless, David
To: Maier, Christi
Cc: Circle, Jeff; Weerakkody, Sunil; Vegel, Anton
Subject: Revised Assumption 22 in Fort Calhoun SERP Package
Date: Tuesday, February 21, 2012 9:38:06 AM

Christi,

Based on comments from APOB, there was some misunderstanding about Assumption 22. Below is the recommended revision from APOB to help correct that misunderstanding. Please send this out to all recipients of the original package.

Thanks,

David

22. **For the estimation of conditional core damage probability (CCDP), a 24-hour mission time was assumed. However, in order to calculate common cause failure of a second circuit breaker fire to start, a vulnerability time of 56 hours was assumed based on the following considerations.**

    Technical Specification 2.7(2)f. permits one of the buses connected to Bus 1A3 or 1A4 to be inoperable for up to 8 hours. Technical Specification 2.7(2), "Modification of Minimum Requirements," requires that with Paragraph f not met:

    > ". . . the reactor shall be placed in hot shutdown within the following 12 hours. If the violation is not corrected within an additional 12 hours, the reactor shall be placed in a cold shutdown condition within an additional 24 hours."

    The analyst noted that licensed operators may decide to cool down the reactor more rapidly than required by Technical Specifications. However, many of the scenarios would require multiple manual system alignments to achieve cold shutdown presenting a potential that reactor cooldown timing would be limited more by manpower available than by license restrictions. Therefore, **for the calculation of common cause failure,** the analyst assumed that the reactor would be in a condition above cold shutdown for 56 hours following a postulated bus fire.

David P. Loveless
Senior Reactor Analyst
U.S. NRC, Region IV

(817) 200-1161

C47

release in entirety

EA Number:
EA-2012-023

Case Information:

EA Number:	EA-2012-023	**Status:**	Open						
Initiator:	REGION IV	**HQ Enf. Specialist:**	Gerald Gulla	**Program Office:**	NRR	**Regional Enf. Specialist:**	Ray Kellar		
Regional Backup Specialist:	Christi Maier	**Regional Division:**	DRS	**Regional Contact:**	Geoff Miller	**Date of Violation:**	06/7/2011		
Continuing Violation?:	Yes	**Enforcement Type:**	SDP	**Discrimination?:**	No	**Referred to DOL?:**	No		
Pending Escalated Action:	Yes	**Timeliness Start Date:**	01/30/2012	**Timeliness Based On:**	Inspection	**Count in Timeliness Report:**	Yes		
License Type:	Operating Reactor	**Small Business?:**	No	**Publicly Available?:**	Yes	**License Number:**	DPR-40		

Short Description:

Preliminary Red Problem dealing with switchgear fire that started in 480 Vac feeder breakers. Three violations associated with Criterion III, Criterion XVI, and License Condition 3.D for Fire Protection Program.

Docket No. 05000285 **License Name:** Omaha Public Power District **Plant Name:** Fort Calhoun

No Outside Organization associated with this case. Click here to add

Facts:

Preliminary Red Problem dealing with switchgear fire that started in 480 Vac feeder breakers. Three violations associated with Criterion III, Criterion XVI, and License Condition 3.D for Fire Protection Program.

Web Summary:

Actions - 1

Panel

Action Information:

Panel Number: 1

Region Participants: Ray Kellar, Tony Vegel, Jeff Clark, Heather Gepford, Christi Maier, David Loveless, Geoff Miller, Rick Deese, Sam Graves, Karla Fuller, Neil O'Keefe, Kris Kennedy, Jeff Josey

OE Participants: Roy Zimmerman, Nick Hilton, Gerry Gulla

Program Office Participants: NRR Jeff Circle, Sunil Weerakkody, Joe Giitter, Tim Kobetz, Rebecca Sigmon, Rani Franovich

Last Updated By: Gulla, Gerald on 03/1/2012

Activities:

OE Approval :

OE Approval: 04/11/2012

Approved by: G Gulla

Last Updated By: Gulla, Gerald on 04/11/2012

Panel Held :

Date: 02/23/2012

Last Updated By: Gulla, Gerald on 03/1/2012

Strategy Form :

Strategy Form Number: 1

Considering Sanctions (CP or Enforcement Order)?: No

Violation Number: A

Violation: Part 50, Appendix B, Criterion III - QA Criteria, Design Control

Date of Violation: 06/7/2011

Specific Issue (Violation) Description:

C48

The failure to ensure that design changes were subject to design control measures commensurate with those applied to the original design; and, that measures were established to assure that applicable regulatory requirements and the design basis, for those safety-related structures, systems, and components were correctly translated into specifications, drawings, procedures, and instructions. From November 2009 to June 7, 2011, the licensee failed to ensure that design changes were subject to design control measures commensurate with those applied to the original design; failed to assure that applicable regulatory requirements and the design basis for those safety-related structures, systems, and components to which this appendix applies were correctly translated into drawings, procedures, and instructions; and failed to ensure that these measures include provisions to assure that appropriate quality standards were specified and included in the design documents. Specifically, design reviews, work planning and instructions for a modification to install new 480 Vac load center breakers failed to ensure that the cradle adapter assemblies had low resistance connections with the switchgear bus bars by establishing a proper fit and requiring low resistance measurements to assure that design basis requirements were maintained.

SDP?: Yes

NOV: Yes

Wrongdoing Information

Wrongdoing: No

Escalated Action

SL/Significance: Red

CP?: No CP

Keywords: Fire Protection

Enforcement Discretion?: No

Decisions Reached

Next Action: Choice Letter

Remarks:

The panel agreed on the 3 violations. There was some discussion regarding the following: 1). the language used in the choice letter regarding the finding color. It was decided to use the word RED vice Greater than Green in the letter. 2). assumption 22, the time until cold shutdown. For CCDP a 24-hour mission time was assumed. However, to calculate a common cause failure of a second circuit breaker fire to start used 56 hours. 3). assumption 30, seismic event. A seismic event could result in the failure of the breaker/breaker cradle interface and/or bolted bus bars in a manner similar to the fire that occurred. 4). the possibility of a civil penalty due to the high significance of the finding. The panel decided not to pursue it because there were no actual consequences as described in the Enforcement Policy. Send the choice letter to HQ for a quick review.

Last Updated By: Gulla, Gerald on 03/9/2012

Strategy Form :

Strategy Form Number: 2

Considering Sanctions (CP or Enforcement Order)?: No

Violation Number: B

Violation: Part 50, Appendix B, Criterion XVI - Corrective Action Program

Date of Violation: 06/7/2011

Specific Issue (Violation) Description:

Failure to establish measures to assure that a significant condition adverse to quality was promptly identified and corrected, and measures taken to preclude repetition. From May 22, 2008, to June 7, 2011, the licensee failed to assure that the case of the significant condition adverse to quality was determined and take corrective actions to preclude repetition. Specifically, the licensee failed to ensure that their preventative maintenance program for the safety-related 480 Vac electrical power distribution system was adequate to ensure proper cleaning of conductors, proper torquing of bolted conductor or bus bar connections, and adequate inspection for abnormal connection temperatures. In 2008, the licensee identified that preventative maintenance procedure EM-PM-EX-1200, "Inspection and Maintenance of Model AKD-5 Low Voltage Switchgear," was less than adequate as a result of a root cause analysis for the failure of bus-tie breaker BT-1B3A to close on demand and loss of bus 1B3A. The licensee categorized this failure as a significant condition adverse to quality. The analysis concluded, in part, that breaker BT-1B3A had high resistance connections, which occurred as a result of both procedure deficiencies and inadequate implementation, resulting in the failure to remove dirt and hardened grease from electrical contacts. The licensee implemented corrective actions to address these procedural deficiencies; however, the corrective actions were inadequate to prevent high resistance connections in load center 1B4A due to the presence of hardened grease and oxidation.

SDP?: Yes

NOV: Yes

Wrongdoing Information

Wrongdoing: No

Escalated Action

SL/Significance: Red
CP?: No CP
Keywords: Fire Protection
Enforcement Discretion?: No

Decisions Reached

Remarks:

Same remarks as in strategy form 1.

Last Updated By: Gulla, Gerald on 03/9/2012

Strategy Form :

Strategy Form Number: 3
Considering Sanctions (CP or Enforcement Order)?: No
Violation Number: C
Violation: License Condition
Date of Violation: 06/7/2011
Specific Issue (Violation) Description:

Failure to ensure that the electrical protection and physical design of the 480 Vac electrical power distribution system provided the electrical bus separation required by the fire protection program. From November 2009, to June 7, 2011, the licensee failed to implement and maintain in effect all provisions of the approved Fire Protection Program. Specifically, the licensee failed to ensure that design reviews for electrical protection and train separation of the 480 Vac electrical power distribution system were adequate to ensure that a fire in load center 1B4A would not adversely affect operation of redundant safe shutdown equipment in load center 1B3A, such that one train of systems necessary to achieve and maintain hot shutdown conditions were free of fire damage as required by the fire protection program. Combustion products from the fire in load center 1B4A migrated across normally open bus-tie breaker BT-1B4A into the non-segregated bus duct, shorting all three electrical phases. The non-segregated bus ducting electrically connected load center 1B4A with the Island Bus 1B3A-4A and, through normally closed bus-tie breaker BT-1B3A, to the redundant safe shutdown train.

SDP?: Yes
NOV: Yes

Wrongdoing Information

Wrongdoing: No

Escalated Action

SL/Significance: Red
CP?: No CP
Keywords: Fire Protection

Decisions Reached

Remarks:

Same remarks as in strategy form 1.

Last Updated By: Gulla, Gerald on 03/9/2012

View legacy data for EA-2012-023

Howell, Art

From: Blount, Tom
Sent: Friday, November 09, 2012 5:02 PM
To: Howell, Art
Cc: Werner, Greg; Collins, Elmo
Subject: Codes for review

Art – you have indicated to me on multiple occasions that there is a need to review SONGS models other than the S/G Thermal Hydraulic model used by the licensee as part of the corrective actions (ATHOS). In our hallway conversation you re-iterated that we expected NRR to look at the AVB- Tube support code (?) and another modeling code, which I cannot recall.
Please refresh me on what models/codes you are thinking we need NRR/DSS to review because I have not communicated that to them as of this time. In part because I did not understand that to be different from what we had previously been seeking from them. After our conversation I think there is something I have missed in communicating.

Thanks,
Tom

Tom Blount
(Acting) Dir DRS R-IV
817-200-1146

C49