UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE ROAD, SUITE 210
LISLE, IL 60532-4352

November 29, 2011

EA-11-241

Mr. Anthony Vitale
Vice-President, Operations
Entergy Nuclear Operations, Inc.
Palisades Nuclear Plant
27780 Blue Star Memorial Highway
Covert, MI 49043-9530

SUBJECT: PALISADES NUCLEAR PLANT, NRC INSPECTION REPORT 05000255/2011016; PRELIMINARY WHITE FINDING

Dear Mr. Vitale:

On October 28, 2011, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Palisades Nuclear Plant. The enclosed report documents the results of this inspection, which were discussed on October 28, 2011, with you and other members of your staff.

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

This report documents a finding that has preliminarily been determined to be White or a finding with low-to-moderate increased safety significance. As documented in Section 4OA5 of this report, a safety-related service water pump (P-7C) failed on August 9, 2011, due to intergranular stress corrosion cracking on coupling #6. This event was a repeat of a September 29, 2009, failure on the same pump, coupling #7, due to the same cause. Based on our assessment of available information, the pump coupling susceptibility to intergranular stress corrosion cracking was introduced in 2007 when a design change to the coupling material was performed. This finding was assessed based on the best available information, including influential assumptions, using the applicable Significance Determination Process (SDP).

Upon identification of this issue, you declared the pump inoperable. As part of the restoration process, all the P-7C couplings and two portions of the pump shaft were replaced. The pump was returned to service on August 12, 2011, within the time allowed by the Technical Specifications action statement. In addition, the couplings of all three service water pumps have been replaced with a material that is less susceptible to intergranular stress corrosion cracking. Because of the actions taken, no current safety concern exists.

This finding is also associated with two apparent violations of NRC requirements which are being considered for escalated enforcement action in accordance with the NRC Enforcement Policy. The current Enforcement Policy can be found at the NRC's Web site at http://www.nrc.gov/reading-rm/doc-collections/enforcement.

In accordance with Inspection Manual Chapter (IMC) 0609, we intend to complete our evaluation using the best available information and issue our final determination of safety significance within 90 days of the date of this letter. The SDP encourages an open dialogue between the NRC staff and the licensee; however, the dialogue should not impact the timeliness of the staff's final determination.

Before the NRC makes its enforcement decision, we are providing you an opportunity to either: (1) present to the NRC your perspectives on the facts and assumptions used by the NRC to arrive at the finding and its significance at a Regulatory Conference or (2) submit your position on the finding to the NRC in writing. If you request a Regulatory Conference, it should be held within 30 days of the receipt of this letter and we encourage you to submit supporting documentation at least one week prior to the conference in an effort to make the conference more efficient and effective. If a conference is held, it will be open for public observation. The NRC will also issue a press release to announce the conference. If you decide to submit only a written response, such submittal should be sent to the NRC within 30 days of the receipt of this letter. If you decline to request a Regulatory Conference or to submit a written response, you relinquish your right to appeal the final SDP determination; in that, by not doing either, you fail to meet the appeal requirements stated in the Prerequisite and Limitation Sections of Attachment 2 of IMC 0609. In addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the Palisades Nuclear Plant.

Please contact John Giessner at (630) 829-9619 and in writing within 10 days of the date of this letter to notify the NRC of your intended response. If we have not heard from you within 10 days, we will continue with our significance determination and enforcement decision. The final resolution of this matter will be conveyed in separate correspondence.

Since the NRC has not made a final determination in this matter, no Notice of Violation is being issued for this inspection finding at this time. Please be advised that the number and characterization of the apparent violation described in the enclosed inspection report may change as a result of further NRC review.

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 NRC's Agencywide Documents Access and Management System (ADAMS), accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html.

Sincerely,

/RA/

Steven West, Director
Division of Reactor Projects

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

Enclosure: Inspection Report 05000255/2011016
w/Attachments:
1. SDP Phase 3 Analysis
2. Supplemental Information

cc w/encl: Distribution via ListServ

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-255
License No:	DPR-20
Report No:	05000255/2011016
Licensee:	Entergy Nuclear Operations, Inc.
Facility:	Palisades Nuclear Plant
Location:	Covert, MI
Dates:	October 4, 2011, through October 28, 2011
Inspectors:	J. Ellegood, Senior Resident Inspector T. Taylor, Resident Inspector D. Betancourt-Roldan, Reactor Engineer J. Jandovitz, Project Engineer N. Valos, Senior Reactor Analyst D. Passehl, Senior Reactor Analyst
Approved by:	John B. Giessner, Chief Branch 4 Division of Reactor Projects

Enclosure

TABLE OF CONTENTS

SUMMARY OF FINDINGS

Inspection Report 05000255/2011016; 10/04/21011 – 10/28/2011; Palisades Nuclear Plant; Other Activities.

This report covers the review of the failure of a service water pump (P-7C) and whether the corrective actions taken after a 2009 failure of the same pump were adequate to prevent recurrence. The inspectors identified a finding with a preliminary significance of White and two associated apparent violations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Initiating Events

- Preliminary White. A self revealed finding with a preliminary low to moderate safety significance and two associated apparent violations of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions," and Criterion III, "Design Control," was self-revealed on August 9, 2011, due to the licensee's failure to prevent recurrence of a significant condition adverse to quality. Specifically, on September 29, 2009, coupling #7 on service water pump P-7C failed due to intergranular stress corrosion cracking (IGSCC). The corrective actions taken to prevent recurrence did not consider all critical factors to prevent or minimize IGSCC from recurring. On August 9, 2011, coupling #6 on service pump P-7C failed due to IGSCC. In addition, in 2007, when the licensee implemented a design change to the coupling material, the licensee failed to reasonably address the factors to reduce susceptibility of the 416 stainless steel couplings to IGSCC. This issue was entered into the licensee's corrective action program (CAP) as CR-PLP-2011-03902. Long term corrective actions included replacing all couplings in the three service water pumps with couplings made of a material that was less susceptible to intergranular stress corrosion cracking.

  This finding was determined to be more than minor because the finding was associated with the Initiating Events Cornerstone attribute of Design Control and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during power operation. Specifically, as a result of the performance deficiency, on August 9, 2011, pump P-7C failed during normal operation. The inspectors performed a Phase 1 SDP evaluation and determined that a Phase 2 evaluation was required because this finding contributed to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. The inspectors then performed a Phase 2 evaluation using the pre-solved SDP worksheets for Palisades and determined that this finding screened as Yellow. Due to inherent conservatisms in the Phase 2 analysis, the RIII Senior Reactor Analysts performed a Phase 3 SDP analysis. The results of the Phase 3 SDP evaluation concluded that this finding was preliminarily determined to be White. The finding has a cross-cutting aspect in the area of Problem Identification and Resolution, Operating Experience, because the licensee failed to take into consideration significant operating experience from as early as 1993 and as late as 2010 that linked IGSCC susceptibility of 410 and 416 stainless steels to temper embrittlement (P.2 (b)). (Section 4OA5.1)

B. Licensee-Identified Violations

No violations were identified.

REPORT DETAILS

4. OTHER ACTIVITIES

4OA5 Other Activities

.1 (Closed) Unresolved Item 05000255/2011012-01; Adequacy of Service Water Pump Couplings

a. Inspection Scope

The inspectors reviewed the circumstances surrounding the August 9, 2011 failure of the safety-related service water (SW) pump P-7C. The inspectors reviewed the licensee's root cause evaluation, design documentation, and the metallurgical analyses performed by an independent laboratory on the failed coupling material.

b. Findings

Introduction: An finding having a preliminary significance of White with two apparent violations of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions," and of Criterion III, "Design Control," was self-revealed on August 9, 2011, due to the licensee's failure to prevent recurrence of a significant condition adverse to quality. In addition, in 2007, when the licensee implemented a design change to the coupling material, the licensee failed to completely consider the properties of 416 stainless steel (SS), a material susceptible to inter granular stress corrosion cracking (IGSCC).

Description: In NRC Inspection Report 05000255/2011012, issued on October 4, 2011, the NRC documented an Unresolved Item (URI) regarding the failure of SW pump P-7C that occurred on August 9, 2011. Specifically, the inspectors identified concerns related to the adequacy of the procurement of the replacement coupling after a 2009 coupling failure in the same pump; whether operating experience was adequately incorporated into design specifications; and whether the corrective actions taken after the 2009 failure were adequate to prevent recurrence. At the conclusion of the inspection, metallurgical analyses of the failed coupling and root cause analysis had not been completed and, for that reason, the issue was documented as unresolved pending NRC review of the completed evaluation.

Upon completion of the site's metallurgical (by an independent laboratory) and root cause analyses, the inspectors reviewed the results. The analyses performed showed that the cause for the failure of coupling #6 (made of ASTM A582 Type 416 SS) on August 9, 2011, was IGSCC. This was a repeat of a previous event that occurred on September 29, 2009, in which improper tempering led to the failure of coupling #7 of the same pump (P-7C) due to IGSCC.

In September 2009, Pump P-7C experienced a failure of coupling #7 that rendered the pump inoperable. Subsequent metallurgical analysis determined the coupling's hardness (between 37-41 Rockwell C (Rc) or HRC) was significantly higher than the design specification (28 to 32 Rc) and that the failure was caused by IGSCC. This failure was entered into the licensee's corrective action program and meets the licensee's definition of a significant condition adverse to quality (SCAQ), and the licensee indicated that it was a SCAQ. The licensee's general criteria for an SCAQ are failures which have, or could result in, a significant degradation or challenge to nuclear safety. The licensee's root cause for the failure was that the vendor had poor quality control in place resulting in a coupling with out-of-specification hardness being provided to the licensee. The licensee's corrective actions in 2009 included replacing all the couplings on SW pump P-7C with couplings of the same design material and hardness criteria. Corrective actions focused on hardness with no other evaluation or corrective actions related to the susceptibility of the material to IGSCC. The failure in 2011, on an adjacent coupling in the same pump, also due to IGSCC, also meets the licensee's definition of a SCAQ. The inspectors determined the SCAQ designation was reasonable based on conditions that existed in 2009 and 2011.

Hardness test results of couplings removed from pump P-7C on August 10, 2011, indicated that these couplings were also not within the hardness specification. Specifically, three of the couplings had at least one test location where the hardness measurement was higher than the licensee's design specification criteria of a maximum of 32 Rc. The highest hardness recorded on the failed 2011 coupling was 33.6 Rc (surface) and 32.7 Rc (through thickness). A third party, independent consultant contracted in 2011 by the licensee reported that the hardness of the material, for values slightly outside of the 28-32 Rc band, was not indicative of the material's susceptibility to IGSCC and could be attributed to the equipment and methodology used to determine hardness. The consultant stated that the susceptibility of the material to IGSCC depended on material toughness which would be affected by the tempering method and temperatures used. For many materials, hardness can provide a reasonable assessment of material embrittlement. However, for 416 stainless and similar steels, heat treatment methods can impact material toughness while yielding acceptable hardness data. Therefore, resistance to IGSCC for 416 stainless steel is best assessed by measuring the material's toughness, a property not required by the site's design criterion nor measured by the licensee's vendor that provided the couplings. Based on this information, the inspector determined that the couplings not adhering to procurement specifications was a performance deficiency, but that the issue was minor since, as specified above, the available information indicates that values slightly outside of the 28-32 Rc band are not indicative of a significant increase in the material's susceptibility to IGSCC.

The inspectors then proceeded to evaluate whether the corrective actions taken after the 2009 failure were adequate to prevent recurrence. The inspectors' review of the root cause evaluation for the 2009 coupling failure revealed that the licensee focused on the material hardness being outside the required specifications, and did not evaluate the effects of toughness or heat treatment on the susceptibility of the couplings to IGSCC. The inspectors determined IGSCC requires three items for occurrence: a susceptible material, tensile stress, and a corrosive environment. The Palisades SW pumps met all three criteria in that 416 SS is susceptible to IGSCC at low toughness values, the vertical SW pumps had tensile stresses exceeding the level necessary to initiate and grow flaws with IGSCC, and the chloride levels in Lake Michigan of about 10 parts per million (ppm) combined with an oxygen rich environment due to periodically wetting and aerating the coupling, met the threshold for a corrosive environment.

In both the root cause and engineering design, the licensee failed to consider the heat treatment methodologies used by the vendor. The vendor's method of heat treatment included hardening of the component through a high heating and quenching process. In some cases, the process resulted in out-of-specification hardness and the vendor would re-temper the coupling. One phenomenon that can adversely affect metallurgical characteristics is temper embrittlement. For 416 stainless steel, temper embrittlement occurs between ~ 700 – 1050 degrees Fahrenheit (°F) based on technical literature, resulting in a material that has a significantly lower fracture toughness. This loss of toughness increases susceptibility to IGSCC and would not be revealed by measuring material hardness. The couplings in question were tempered in the 1025 – 1090 °F range. Information regarding temper embrittlement and toughness can be found in operating experience from as early as 1993. Information Notice 93-68 discusses IGSCC and material failures on similar steels (410 SS) in raw water systems at nuclear plants. Another example of more recent operating experience includes IN 2007-05, which discussed a 2005 Columbia Generating Station shaft failure due to tempering embrittlement and a 2004 Perry Station failure of 416 SS couplings due to IGSCC. The licensee did not take this operating experience into account.

The 2009 root cause did not address the toughness of the material and focused only on hardness. A report written by an independent group hired by the licensee, “Additional Review of Palisades Service Water Pump Couplings,” Revision 0, dated March 2011, stated that hardness, alone, is not a good indicator of IGSCC susceptibility. The licensee requested this report between the first and second Palisades’ coupling failures, based on a similar failure of a 410 SS service water pump coupling which occurred at Prairie Island in 2010. While 410 SS is generally considered to be less susceptible to IGSCC than 416 SS, it is otherwise very similar in regard to material properties. The following are excerpts from the report:

> “…[the Prairie Island metallurgist’s] conclusion that IGSCC susceptibility and ***material toughness*** [highlighted in original report] was properly placed, however, the correlation between IGSCC susceptibility and a hardness of HRC28-HRC32 was not generally correct: That is, Type 410 Stainless Steel can have a hardness of HRC32 and be reasonably tough with good IGSCC resistance, while the same material at the same hardness level can have very low toughness and relatively poor IGSCC resistance if the heat treatment that produces the final hardness produces temper embrittlement, a condition that will not be indicated by the hardness.”
>
> “SCC susceptibility correlates well with toughness; much better than with hardness.” That is, the tougher the material, the more SCC resistant.

In addition, the 2009 root cause did not address the adequacy of the material for the environment to which it was subjected, or the use of other parameters, which would provide clues to its susceptibility to IGSCC based on the current heat treatment methods, even though there was enough expertise and operating experience which indicated this should be done. The shaft coupling material for P-7 A/B/C had been changed from carbon steel to 416 SS under Engineering Change (EC) 5000121762, in December of 2007. The EC mentioned that the ASTM A582 Type 416 SS was chosen due to its material strength, wear resistance and corrosion resistance. The stainless steel couplings were put into service on P-7C in June 2009. However, operating experience showed that the type of SS selected is susceptible to IGSCC under certain conditions. Therefore, the licensee failed to verify that the material was adequate for the environment and working conditions for which it would be subjected. As a result, the licensee failed to identify and evaluate a new failure mechanism, which was introduced into the system in the form of IGSCC. In summary, the corrective actions from the 2009 event focused on the hardness of the material, but it did not address the underlying issue that led to the susceptibility to IGSCC, and ultimately led to the 2011 failure.

As part of the inspection, the inspectors also reviewed the licensee's root cause completed in September of 2011, which indicated, as one root cause, that both the 2009 and 2011 failures occurred due to IGSCC. The report specified that: "the coupling material is a quenched and tempered 416 martensitic SS with low toughness properties. This makes it particularly susceptible to IGSCC when subjected to the tensile stress and a corrosive environment (due to the presents [sic] of chlorides)." In addition, the September 2011 report concluded that the "root cause evaluation conducted after the failure [in 2009] did not sufficiently investigate the base material properties of 416SS. Specifically, corrosion in the Lake Michigan water environment and the toughness properties of the material were not investigated." The licensee's other root cause (for the 2011 event) was that, in 2007, the engineering group specified the wrong SS alloy for use in a chloride environment. The inspectors concluded the root causes were aligned with the inspectors' assessment of the event.

As part of the corrective actions for the 2011 event, the licensee changed the coupling material and replaced the couplings on each of the three pumps P-7 A/B/C with a new type of stainless steel (17-4 precipitation hardened (PH)), which is less susceptible to IGSCC.

<u>Analysis</u>: The inspectors determined that the licensee's failure to prevent recurrence of a safety-related service water pump failure on August 9, 2011 (P-7C, coupling #6) due to IGSCC was a performance deficiency and subject to the reactor SDP. This failure did not meet regulatory requirements and was within the licensee's ability to identify and correct. On September 29, 2009, coupling #7 on service water pump P-7C failed due to IGSCC, a significant condition adverse to quality. The corrective actions taken to prevent recurrence did not consider all the critical factors to minimize or prevent IGSCC from recurring. In addition, in 2007, when the licensee implemented a design change to the coupling material, the licensee failed to reasonably address the factors to reduce susceptibility of the 416 stainless steel couplings to IGSCC. The inspectors screened the performance deficiency in accordance with Inspection Manual Chapter (IMC) 0612, Appendix B, "Issue Screening." The performance deficiency was determined to be more than minor because the finding was associated with the Initiating Events Cornerstone attribute of Design Control, and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during power operation. Specifically, as a result of the performance deficiency, on August 9, 2011, pump P-7C failed during normal operation.

The inspectors evaluated the finding in accordance with Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of findings," Table 4a for the Initiating Events Cornerstone. The inspectors answered "Yes" to the screening question for Transient Initiators "Does the finding contribute to both the likelihood of a reactor trip AND the likelihood that mitigation equipment or functions will not be available?," since an increased failure-to-run rate and a failure of a service water pump both increases the frequency of a loss of service water initiating event and increases the probability that the service water system will not be available following an initiating event. Therefore, a Phase 2 Significance Determination Process (SDP) evaluation was performed using IMC 0609, Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations.” The Senior Reactor Analysts (SRAs) performed a Phase 2 evaluation using the pre-solved SDP worksheets for Palisades and determined that this finding screened as Yellow. Due to inherent conservatisms in the Phase 2 analysis, a Phase 3 SDP analysis was performed by the SRAs.

The calculations for the Phase 3 SDP analysis are included in Attachment 1 of this document. The conclusion of the Phase 3 analysis was an estimated change in CDF of 5.4E-6/year or WHITE.

This finding has a cross-cutting aspect in the area of Problem Identification and Resolution, Operating Experience, because the licensee failed to take into consideration significant operating experience from as early as 1993 and as late as 2010 that linked IGSCC susceptibility with material toughness and shaft failures due to temper embrittlement (P.2 (b)).

<u>Enforcement</u>: During the inspection, the inspectors identified two apparent violations of NRC requirements:

- Title 10 CFR, Part 50, Appendix B, Criterion XVI, “Corrective Actions,” states, in part, that “In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective actions taken to preclude repetition.”

  An apparent violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Actions,” has been identified as it appears that the licensee failed to prevent the recurrence of a significant condition adverse to quality. Specifically, on September 29, 2009, coupling #7 of P-7C failed due to IGSCC. The licensee’s action to prevent recurrence did not consider all critical factors to prevent IGSCC from recurring. On August 9, 2011, coupling #6 of P-7C also failed due to IGSCC. Therefore, the corrective actions from the first event failed to prevent recurrence.

- Title 10 CFR, Part 50, Appendix B, Criterion III, “Design Control,” requires, in part, that the design control measures shall be established for the selection and review for suitability of application of materials, parts, equipment and processes that are essential to the safety-related functions of the structures, systems and components.

  An apparent violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” has been identified as it appears the licensee failed to select materials suitable for the safety-related function of the service water pump couplings. Specifically, in December 2007, the licensee modified the design of the service water pump couplings to change the material from carbon steel to 416 stainless steel. The licensee failed to verify that the material was adequate for the environment and working conditions for which it would be subjected. As a result, the licensee failed to identify and evaluate a new failure mechanism, which was introduced into the system in the form of IGSCC.

This issue was entered into the licensee's corrective action program as CR-PLP-2011-03902. The finding and associated apparent violations of 10 CFR, Part 50, Appendix B, Criterion XVI and Criterion III, are of preliminary White significance pending completion of the final significance determination **(AV 05000255/2011016-01, Failure to Prevent Recurrence of a Significant Condition Adverse to Quality)**.

4OA6 Management Meetings

.1 Exit Meeting Summary

On October 28, 2011, the inspectors presented the inspection results to Mr. A. Vitale, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

ATTACHMENTS:
1. SDP PHASE 3 ANALYSIS
2. SUPPLEMENTAL INFORMATION

SDP PHASE 3 ANALYSIS

To calculate the exposure time, Section 2.1 of Volume 1 of the Risk Assessment of Operational Events (RASP) Handbook was used. The RASP Handbook states that the exposure time is the duration period of the failed or degraded structure, system, or component (SSC) being assessed that is reasonably known to have existed and includes repair time. For the P-7C pump, the exposure time was one year (the maximum allowable time used in risk analyses), based upon the new stainless steel material for the couplings for all three SW pumps being in place since at least mid-May 2010. The P-7C pump failed on August 9, 2011, at 1201 hours and was returned to service on August 12, 2011, at 0309 hours following successful post-maintenance surveillance testing. Thus, the repair time was approximately 63 hours. There is no recovery credit in this analysis.

This analysis divided the exposure time into two segments:

- The exposure time with the P-7C SW pump not failed (for approximately one year), but with an increased failure-to-run (FTR) rate for all three SW pumps.
- The exposure time when the P-7C SW pump was failed (approximately 63 hours), with an increased FTR rate for SW pumps P-7A and P-7B.

The SRA used the Palisades Standardized Plant Analysis Risk (SPAR) model (version 8.17 dated June 20, 2011), and the SAPHIRE 8 version 8.0.17 software.

The Palisades SPAR model was modified using the Events and Conditions Assessment (ECA) workspace with the following changes:

- A revised FTR rate for the three SW pumps was obtained using statistical analysis, i.e., a Bayesian update with a Jeffreys non-informative prior. The two observed failures of the SW pumps over a total run-time of 40509 hours for the three SW pumps (since the new stainless steel couplings were installed until the failure of the P-7C pump on August 9, 2011) was used. The revised FTR rate for the three SW pumps was 6.17E-5/hour.

- A revised initiating event frequency (IEF) for a loss of service water (IE-LOSW) was obtained based on an approximate method recommended by Idaho National Laboratory (INL). To estimate a new IEF for the LOSW event, the existing SW system fault tree was solved with the nominal SW pump FTR failure rate (3.9E-6/hour), and then again with the new FTR rate (6.17E-5/hour). The ratio of the SW system unavailability with the new rate to that of the system unavailability with the old rate was then calculated. The IEF for the LOSW was then increased by this ratio.

- For the exposure time of approximately 1 year with all three SW pumps having an increased FTR rate, the IEF for the LOSW event increased by a factor of 3.23 (from 2.50E-4/year to 8.06E-4/year).

- For the exposure time from August 9, to 11, 2011, of 63 hours (with the P-7C SW pump failed - True), and with an increased FTR rate for P-7A and P-7B SW pumps to 6.17E-5/hour, the IEF for the LOSW event increased by a factor of 1590 (from 2.50E-04/year to 0.40/year).

A common cause failure (CCF) potential associated with the performance deficiency for all three SW pumps was assumed. Consistent with the RASP Handbook, a component failure should only be modeled as an independent failure if the cause is well understood and there is no possibility that the same circumstance exists in other components in the same common-cause component group. Based on this, it was assumed that there was a CCF potential associated with all three SW pumps.

The change in core damage frequency (CDF) risk was evaluated for each of the two segments of the exposure time and the results were added together to get a total internal events CDF risk.

<u>Case 1</u>: P-7C SW pump <u>not</u> failed (approximately 1 year), but with an increased FTR rate for all three SW pumps.

For the exposure time of approximately 1 year with all three SW pumps having an increased FTR rate of 6.17E-5/hour and an increased IEF of 8.06E-4/year, the CDF was calculated to be 7.6E-7/year.

<u>Case 2</u>: P-7C SW pump failed (approximately 63 hours), and with an increased FTR rate for P-7A and P-7B SW pumps.

For the exposure time from August 9, to 11, 2011, of 63 hours with the P-7C SW pump failed and with an assumed increase in the FTR rate for P-7A and P-7B SW pumps of 6.17E-5/hour and an increased IEF to 0.40/year, the CDF was calculated to be 3.9E-6/year.

The total internal events CDF is the sum of the two CDFs calculated above or 4.7E-06/year.

The dominant sequences involved a loss of service water system initiating event, failure of reactor coolant pump (RCP) seal cooling, failure of SW system recovery, and containment cooling failure cutsets.

Since the total estimated change in core damage frequency was greater than 1.0E-7/year, IMC 0609, Appendix A, Attachment 3, was used to assess external event contributions.

The fire risk contribution was estimated using information from the licensee's Individual Plant Examination for External Events (IPEEE), Revision 1, dated May 22, 1996. From Section 4.0.4 of the IPEEE, the core damage frequency from fires is 3.31E-5/year.

In Table 4.12-2, "Risk Significant Operator Actions for the Fire Analysis," a Risk Achievement Worth (RAW) value for "failure to align alternate suction source to auxiliary feedwater (AFW) upon depletion of the Condensate Storage Tank (CST)" is given the value of 3.6. The increase in the failure probability of the SW system (used as a suction source to the AFW system) due to the performance deficiency was calculated to be 3.44E-3.

An estimate of the CDF for the fire risk was obtained as:

CDF(fire) = [RAW – 1] x [Increase in Failure Probability of SW] x [CDF for fires]
= [3.6 – 1] x [3.44E-3] x [3.31E-5/year]
= 3.0E-7/year

The total estimated CDF from fires is thus 3.0E-7/year.

The seismic risk contribution was estimated using information from the licensee's IPEEE, Revision 1, dated May 22, 1996. From Table 3.6-3 of the IPEEE, the total core damage frequency from Class IA and Class IB seismic events was 6.16E-6/year.

Failure of secondary heat removal requires the loss of the AFW system. AFW pumps P-8A and P-8B take suction from the fire protection system (FPS) after the condensate storage tank (CST) is depleted. AFW pump P-8C is the only one of the three AFW pumps that can take suction from the SW system after the CST is depleted. The failure rate of AFW pump P-8C is proportional to the loss of secondary heat removal during a seismic event, and thus is proportional to the total core damage frequency from Class IA and Class IB seismic events. Per Section 3.6.5.3.3 of the IPEEE, there are two dominant random event groups that contribute to the failure of AFW pump P-8C which gives a failure rate of 6.01E-2/year for AFW pump P-8C.

However, with the performance deficiency associated with the SW pump couplings, the increase in the IEF for a LOSW event represented an additional failure rate for AFW pump P-8C. With the performance deficiency, the total failure rate for AFW pump P-8C was 6.35E-2/year (6.01E-2/year + 3.44E-3/year). The fractional increase in the failure rate for AFW pump P-8C due to the performance deficiency was 5.7 percent.

An estimate of the CDF for the seismic risk was obtained as follows:

CDF(seismic) = [Fractional increase in failure rate for AFW pump P-8C] x [CDF for Class IA and Class IB seismic events]
= [0.057] x [6.16E-6/year]
= 3.5E-7/year

The total estimated CDF from seismic events is thus 3.5E-7/year.

Internal flood risk contributions were screened using IMC 0609, Appendix A, Table 3.1, Plant Specific Flood Scenarios. The guidance lists SSCs important to internal flooding and there are no SSCs listed for Palisades.

The total estimated delta CDF for external events is obtained by summing the contributions from the fire risk (3.0E-7/year) and the seismic risk (3.5E-7/year) or 6.5E-7/year.

The total estimated delta CDF is the sum of the internal events contribution (4.7E-6/year) and the external events contribution (6.5E-7/year) or 5.4E-6/year.

The SRAs used IMC 0609 Appendix H, “Containment Integrity Significance Determination Process” to determine the potential risk contribution due to LERF. Palisades Nuclear Plant is a 2-loop Combustion Engineering Pressurized Water Reactor (PWR) with a large, dry containment. Sequences important to LERF include steam generator tube rupture events and inter-system loss of coolant accident (LOCA) events. These were not the dominant core damage sequences for this finding and thus the risk significance due to LERF was evaluated to be of very low safety significance.

In summary, the conclusion of the Phase 3 analysis was an estimated change in core damage frequency of 5.4E-6/year (WHITE). The licensee has not yet provided the results of a risk evaluation for the finding.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

A. Vitale, Entergy, Site Vice President
D. Hamilton, General Manger Plant Operations
A. Blind, Engineering Director
O. Gustafson, Licensing Manager
D. Corbin, Acting Operations Manager
J. Haumersen, Systems Engineering Manager

Nuclear Regulatory Commission

J. Giessner, Chief, Reactor Projects Branch 4
M. Chawla, Project Manager, NRR

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000255/2011016-01	AV	Failure to Prevent Recurrence of a Significant Condition Adverse to Quality (Section 4OA5.1)

Closed

05000255/2011012-01	URI	Adequacy of Service Water Pump Couplings (Section 4OA5.1)

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.

Sections 4OA5

- Risk Assessment of Operational Events RASP Handbook; Volume 1 (Internal Events) and Volume 2 (External Events).
- CR-PLP-2009-04519; Root Cause Evaluation Report for Service Water Pump P-7C coupling failure; March 4, 2009,
- CR PLP-2011-03975; Relevant OE Not Considered in CR-PLP-2009-4519 (Root Cause Evaluation for the 2009 Service Water Pump P-7C coupling failure; August 12, 2011,
- CR-PLP-2011-03961; Potential Extent of Condition on Service Water Pump P-7A and P-7B; August 11, 2011,
- CR-PLP-2011-03966; Removed Coupling from P-7C 2011 failure show Out-of Spec hardness; August 12, 2011,
- CR-PLP-2011-03966; Results of 4 Couplings Sent to Consumers Laboratory Services for Hardness Testing; August 12, 2011;
- CR-PLP-2011-03975; Relevant OE not considered in CR-PLP-2009-04519; August 12, 2011,
- EN-LI-102 “Corrective Action Process”, Revision 16,
- CR-PLP-2011-03902; Root Cause Evaluation Report; Revision 0,
- Report No. 1100112.401; Additional Review of Palisades Service Water Pump Couplings; Revision 0, dated March 2011,
- Report No. F11358-R-001; Metallurgical and Failure Analysis of SWS Pump P-7C Coupling #6, Revision 0; October 2011,
- LPI Ref F11358-LR-001; Past Operability Assessment of Service Water Pumps P-7A and P-7B associated with As-found Evaluation of Pump Shaft Couplings – Palisades Nuclear Plant; Revision 0;
- LPI Report No. F11358-R-001; revision 0, dated October 2011.

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access Management System
AFW	Auxiliary Feedwater
AV	Apparent Violation
CAP	Corrective Action Program
CCF	Common Cause Failure
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CST	Condensate Storage Tank
DRP	Division of Reactor Projects
EC	Engineering Change
ECA	Events and Conditions Assessment
FTR	Failure-to-Run
FPS	Fire Protection System
IEF	Initiating Event Frequency
IMC	Inspection Manual Chapter
IN	Information Notice
IPEEE	Individual Plant Examination of External Events
IGSCC	Intergranular Stress Corrosion Cracking
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
PARS	Publically Available Records System
ppm	parts per million
PWR	Pressurized Water Reactor
RASP	Risk Assessment Operational Events
RAW	Risk Achievement Worth
Rc	Rockwell C
RCP	Reactor Coolant Pump
SCAQ	Significant Condition Adverse to Quality
SDP	Significance Determination Process
SPAR	Standardized Plant Analysis Risk
SRA	Senior Reactor Analyst
SS	Stainless Steel
SSC	Structure, System or Component
SW	Service Water
URI	Unresolved Item

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 NRC's Agencywide Documents Access and Management System (ADAMS), accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html.

Sincerely,

/RA/

Steven West, Director
Division of Reactor Projects

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

Enclosure: Inspection Report 05000255/2011016
w/Attachments: 1. SDP Phase 3 Analysis
2. Supplemental Information

cc w/encl: Distribution via ListServ

See Previous Concurrences

DOCUMENT NAME: G:\DRPIII\PALI\PAL 2011 016.docx
☐ Publicly Available ☐ Non-Publicly Available ☐ Sensitive ☐ Non-Sensitive
To receive a copy of this document, indicate in the concurrence box "C" = Copy without attach/encl "E" = Copy with attach/encl "N" = No copy

OFFICE	RIII	EICS	RIII	RIII
NAME	SShah*:dtp	SOrth*	JGiessner	SWest
DATE	11/16/11	11/16/11	11/29/11	11/29/11

OFFICIAL RECORD COPY

Letter to A. Vitale from S. West dated November 29, 2011.

SUBJECT: PALISADES NUCLEAR PLANT, NRC INSPECTION REPORT 05000255/2011016; PRELIMINARY WHITE FINDING

DISTRIBUTION:
Amy Snyder
RidsNrrPMPalisades Resource
RidsNrrDorlLpl3-1 Resource
RidsNrrDirsIrib Resource
Cynthia Pederson
Jennifer Uhle
Steven Orth
Jared Heck
Allan Barker
Carole Ariano
Linda Linn
DRPIII
DRSIII
Patricia Buckley
Tammy Tomczak
ROPreports Resource