Notes

2	Palisades' specifications for the coupling required use of 416 SS and did not require toughness testing nor adequately test for hardness (insufficient sample size). See Table 2. (RC2)
2	Palisades' specifications for the coupling did not involve input/review by a qualified metallurgist. See Table 2. (CC2)
5	When the Prairie Island 2010 OE became available, Palisades did not initiate a condition report to question the suitability of couplings that were in stock. (CC3)
6	Palisades did not take full advantage of operating experience suggesting that 416 SS was susceptible to IGSCC. See Table 2 (CC3)
10	Palisades' system engineer functioned as a design engineer when dealing with HydroAire. See Table 2. (RC2)

TABLE 2 – DETAILED SAFETY CULTURE COMPONENT REVIEW

		Description	CR-PLP-2011-03902
1. Decision-Making		**Licensee decisions demonstrate that nuclear safety is an overriding priority. Specifically (as applicable):**	
2. Resources		**The licensee ensures that personnel, equipment, procedures, and other resources are available and adequate to assure nuclear safety. Specifically, those necessary for:**	
RES	H.2(b)	Training of personnel and sufficient qualified personnel to maintain work hours within working hours guidelines. Palisades' specifications for the coupling did not involve input/review by a qualified metallurgist.	RC_1 - No indication RC_2 - No indication CC_1 - No indication CC_2 - Yes CC_3 - No indication
RES	H.2(c)	Complete, accurate and up-to-date design documentation, procedures, and work packages, and correct labeling of components. Palisades' specifications for the coupling required use of 416 SS and did not require toughness testing nor adequately test for hardness (insufficient sample size).	RC_1 - No indication RC_2 - Yes CC_1 - No indication CC_2 - No indication CC_3 - No indication
3. Work Control		**The licensee plans and coordinates work activities, consistent with nuclear safety. Specifically (as applicable):**	
4. Work Practices		**Personnel work practices support human performance. Specifically (as applicable):**	
5. Corrective Action Program		**The licensee ensures that issues potentially impacting nuclear safety are promptly identified, fully evaluated, and that actions are taken to address safety issues in a timely manner, commensurate with their significance. Specifically (as applicable):**	
6. Operating experience		**The licensee uses operating experience (OE) information, including vendor recommendations and internally generated lessons learned, to support plant safety. Specifically (as applicable):**	
OE	P.2(a)	The licensee systematically collects, evaluates, and communicates to affected internal stakeholders in a timely manner relevant internal and external OE. Palisades did not take full advantage of operating experience suggesting that 416 SS was susceptible to IGSCC.	RC_1 - No indication RC_2 - No indication CC_1 - No indication CC_2 - No indication CC_3 - Yes
OE	P.2(b)	The licensee implements and institutionalizes OE through changes to station processes, procedures, equipment, and training programs. Palisades did not take full advantage of operating experience suggesting that 416 SS was susceptible to IGSCC.	RC_1 - No indication RC_2 - No indication CC_1 - No indication CC_2 - No indication CC_3 - Yes

		Description	CR-PLP-2011-03902
7. Self- and Independent Assessments		The licensee conducts self- and independent assessments of their activities and practices, as appropriate, to assess performance and identify areas for improvement. Specifically (as applicable):	
8. Environment For Raising Concerns		An environment exists in which employees feel free to raise concerns both to their management and/or the NRC without fear of retaliation and employees are encouraged to raise such concerns. Specifically (as applicable):	
9. Preventing, Detecting, and Mitigating Perceptions of Retaliation		A policy for prohibiting harassment and retaliation for raising nuclear safety concerns exists and is consistently enforced in that:	
10. Accountability		Management defines the line of authority and responsibility for nuclear safety. Specifically (as applicable):	
ACC	A.1(a)	(a) Accountability is maintained for important safety decisions in that the system of rewards and sanctions is aligned with nuclear safety policies and reinforces behaviors and outcomes which reflect safety as an overriding priority. Palisades' system engineer functioned as a design engineer when dealing with HydroAire. Management did not recognize the risk associated with this arrangement.	RC_1 - No indication RC_2 - Yes CC_1 - No indication CC_2 - No indication CC_3 - No indication
11. Continuous learning environment		The licensee ensures that a learning environment exists. Specifically (as applicable):	
13. Safety policies		Safety policies and related training establish and reinforce that nuclear safety is an overriding priority in that:	

Stainless Steel – Grade 416

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NTERNAL Operating Experience:

The Paperless Condition Reporting System (PCRS) was utilized to search for similar events involving coupling failures. The search was limited to Palisades' events as similar searches were performed using Autonomy and documented in the External Operating Experience section, which included all of the Entergy stations.
The search criteria did not include any time constraints and specified keywords "pump coupling". The search yielded twenty seven (27) Condition Reports (CRs). All 27 CRs were reviewed and concluded that none were similar to this coupling failure. An additional search, with no time constraints, was performed with the specified key word "service water pump". The search yielded four hundred and ninety (490) Condition Reports. All 490 CRs were reviewed and concluded that none were similar to this coupling failure; however, a number of CRs were classified and evaluated under three broad areas:

- Foreign Material issues in service water bay (16 CRs)
- Service Water Pump degraded monitoring parameters (19 CRs)
- Disassembly/Assembly Service Water Pump parts degradation/discrepancies. (39 CRs)

Other Entergy site's operating experience was searched using the Autonomy system. Various combinations of keywords involving "service water pumps", "couplings", "embrittlement", and "failures" were used. No events were uniquely identified other than those Entergy sites identified in the External Operating Experience section.

EXTERNAL Operating Experience:
Note: Operating Experience denoted with "*" contains information indicating an upper pump coupling was affected.
CR-PLP-2009-04519, "Service Water Pump P-7C Failure to Provide Discharge Pressure" identified operating experience.
The Institute of Nuclear Power Operations (INPO) Equipment Performance and Information Exchange System (EPIX) and Nuclear Plant Reliability Data System (NPRDS) searches were conducted using search terms "shaft" AND "failure" AND "pump" AND "service water" AND NOT "bent" AND "coupling" AND "Layne Bowler" AND "failure near coupling" AND "Service water". These searches yielded 11 OE articles that were analyzed for applicability as documented further in this attachment.

EPIX Failure #164 - River Bend Unit 1, March 23, 2001 (CR-RBS-2001-00403)
Description:
Service Water Cooling pump failure occurred from fatigue failure of its bolting caused by a corrosion induced loss of pre-load. The bolting for two other Service Water Cooling pumps had not started cracking, but had similar but not as severe corrosion damage. Stainless steel was recommended as a replacement for the carbon steel bolting. The relatively worse condition of the failed pump's bolting was contributed to manual addition of sulfuric acid in its bay during early years of Service Water Cooling system operation.

Applicability to Palisades:
Not relevant for this event. However, consideration to coupling material and service water chemistry needs to be considered.

NPRDS (Pumps, eductors) - Indian Point 2, September 22, 1993
Description:
Pump failure was due to the sudden failure of one of the pump couplings. The pumps discharge check valve was subsequently discovered to be not seating properly and leaking by. This leak-by would cause the pump to rotate backwards, and this would produce an excessive start-upload. The coupling failure is attributed to non-ductile fracture because of temper embrittlement of the 410 SST. ASTM A276 Type 410SST minimum mechanical property requirements are adequate for normal pump loads per this standard but could be exceeded under impact or rapid loading conditions. The pump was replaced with a rebuilt pump using couplings with newly developed heat treatment specifications.

Applicability to Palisades:
Elements of this event are similar to Palisades. The coupling failure mechanism has been determined to be an initial Intergranular Stress Corrosion Crack (IGSCC) that developed and propagated to a point where a load-induced brittle failure occurred.

NPRDS (Pumps, eductors) - Beaver Valley 1, September 12, 1991
Description:
Failure of a Byron-Jackson centrifugal pump resulted from a mechanical failure of the shaft coupling, Lehigh University laboratory test results attributed the coupling failure to embrittlement of the 410 SST material due to improper tempering temperatures and the potential impurities in the steel. The defective couplings were replaced with newly purchased safety-related couplings that were tested to ensure acceptability.

Applicability to Palisades:
Elements of this event are similar to Palisades. The coupling failure mechanism has been determined to be an initial Intergranular Stress Corrosion Crack (IGSCC) that developed and propagated to a point where a load-induced brittle failure occurred.

*EPIX Failure #167 - Perry Unit 1, September 1, 2003
Description:
Emergency Service Water (ESW) A pump lost flow after 42 minutes of operation. Follow up investigation found no evidence of a pump or motor transient or any sign of foreign material obstruction in the pump impellers. Disassembly of other pump found the first line shaft coupling sleeve had failed and was found in two pieces inside the pump assembly. Visual inspection of wear marks on the broken coupling sleeve halves indicated the coupling was not centered between the two shafts. This left approximately one inch of the key extending above the coupling during operation.

Applicability to Palisades:
Palisades also had no evidence of a pump or motor transient or any sign of foreign material obstruction in the pump impellers. Nuclear Regulatory Commission Information Notice 2007-05 indicated the coupling failure was attributed to intergranular stress corrosion cracking. IGSCC was identified as a contributor to the Palisades coupling failures.

EPIX #659 - Catawba Unit 1, July 12, 2008 (C-08-04289)
Description:
Root cause of the Catawba event was the use of the coupling sleeve manufactured from a deficiently formed alloy. Specifically, the pump failed due to the heterogeneity of the upper Johnston coupling material. Heterogeneity caused the material to be highly susceptible to intergranular corrosion and cracking (IGSCC) as shown by an accumulation of fine sulfide stringer inclusions along the boundaries of the failed coupling. Use of Martensitic stainless steel, A582 type 416 condition – T, the maximum hardness level of Rockwell C-25 should be specified to minimize the susceptibility of the material to stress corrosion cracking (SCC). Material susceptibility to SCC is border-line in the mid -20s range. The Johnston coupling had a hardness of Rockwell C-28; therefore, the specified hardness of the Johnston coupling is a contributing cause.

Applicability to Palisades:
Chemical analysis of the 2009 failed coupling at Palisades did not identify any issues with material alloy; therefore the heterogeneity of the Catawba coupling is not an area of concern for Palisades.

*INPO Operating Experience Digest 2006-02
Description:
This document discusses Service Water Pump failures identifying specific plant sites that have experienced these failures. Service water pump shaft, coupling, and impeller failures have been identified as a continuing trend of service water mechanical problems. Twelve failures have been reported to the industry from 1998 through 2006, averaging more than one failure per year with several stations having multiple failures. The most frequent cause has been corrosion of the shaft and bolting material. Corrosion has resulted in bolting failure; shaft shearing, impeller and coupling separation, and binding of the impeller to the bowl. Contributing causes include the following:

- Improper heat treatment during manufacture.
- Incorrect bolting and shaft material specification that are more susceptible to intergranular stress corrosion cracking.
- Use of dissimilar metals resulting in galvanic corrosion.
- Pump operation at low speed resulting in resonance vibration fatigue.
- Stray current flow from the cathodic protection system.
- Excessive bearing wear.
- Abnormal changes in lubricating water temperature (high or low) can result in differential rates of thermal expansion leading to binding of internal parts.

Applicability to Palisades:
This OE is applicable to Palisades because of material similarities and indications of intergranular stress corrosion cracking.

Electric Power Research Institute (EPRI)
EPRI web site was searched for documents using the keywords “Service Water Pumps.” Two documents of interest are “NP-7413, Deep Draft Vertical Centrifugal Pump Maintenance and Application Guide” and “Vertical Pump Maintenance Guide Supplement to NP-7413, Deep Draft Vertical Centrifugal Pump Maintenance and Application Guide.” These documents provide an extensive discussion on vertical pump components, material specifications, maintenance, and troubleshooting. A review of

these documents did not reveal any specifics to this particular event, but does provide a validation of maintenance, operation, and engineering practices with accepted industry standards.

ADAMS Search / External OE document review
The NRC Agency-wide Documents Access and Management System (ADAMS) was queried for events involving "Service Water Pumps."

NRC INFORMATION NOTICE 2007-05: VERTICAL DEEP DRAFT PUMP SHAFT AND COUPLING FAILURES, February 9, 2007
Description:
This Information Notice was issued to alert licensees to vertical deep draft pump shaft and coupling failures from intergranular stress corrosion cracking (IGSCC). Service Water Pump shaft failures were experienced at Columbia Generating Station. The metallurgical examination determined that the shaft material, TP410 martensitic stainless steel, was susceptible to tempering embrittlement. Tempering embrittlement reduced the corrosion resistance of the shaft material, thereby, increasing the material's susceptibility to IGSCC. NRC review of Operating Experience records identified at least 23 essential SW pump shaft and coupling failures since 1983 involving more than six different pump manufacturers. Many of these failures involved IGSCC as a primary cause. Other causes of shaft and coupling failures included: misalignment, imbalance, installation errors, and deferred maintenance. Two incidents since 2001, involving IGSCC are:

- IN 2007-05 - Perry experienced SW pump shaft coupling failures due to IGSCC in September 2003 and May 2004.
- ML020920543 - VC Summer experienced SW pump shaft coupling failure during testing due to IGSCC in May 2001.

Applicability to Palisades:
This OE is applicable to Palisades because IGSCC was identified as a contributor to the coupling failures. This OE was reviewed under LO-PLPLO-2007-00059 and was concluded to be non applicable based on the material characteristics of 416 stainless steel. This OE was also used as input into EC5000121762, but the EC did not acknowledge that a fresh water environment should be considered as a potentially corrosive environment.

NRC INFORMATION NOTICE 93-68: FAILURE OF PUMP SHAFT COUPLING CAUSED BY TEMPER EMBRITTLEMENT DURING MANUFACTURE, September 1, 1993
Description:
The U.S. Nuclear Regulatory Commission issued this information notice to alert addressees to problems caused by temper embrittlement of American Iron and Steel Institute Type 410 stainless steel couplings supplied by Byron Jackson. On June 20, 1991, a river water pump shaft coupling at Beaver Valley Nuclear Power Plant, Unit 1, failed during operation when a large section of one end of the coupling broke away from the rest of the coupling. This coupling, which was threaded internally, was used to join two shafts of a Byron Jackson vertical circulator river water pump. During its investigation of the failure, the licensee found that two more couplings from the same pump had cracks. All three of the Unit 1 pump shafts had at least one of the defective couplings. The licensee at Beaver Valley noted that increased vibration levels on pump

1A caused by a worn bearing, pump shaft misalignment or both contributed to the failure. The cause of the Beaver Valley failure was determined in independent laboratory testing as "impact strength of the couplings due to temper embrittlement resulting from improper heat treatment."

Applicability to Palisades:
This is the same event described in the INPO EPIX search. Elements of this event are similar to Palisades. Palisades Service Water pump couplings are made from 416 SS. 410 SS and 416 SS have similar chemical compositions and properties.

ML020930345 Indian Point 2 Date: 06/30/94
Description:
June 29, 1994, a service water pump failed to develop discharge pressure during a pump start. A contributing cause was the low impact resistance of the material used. There have been three other service water pump failures at Indian Point 2. In August 1993, two pumps failed due to sudden impact loading caused by foreign object ingestion. In September 1993, a pump failure was attributed to impact loading caused by reverse flow through the check valve during startup. In all cases, a contributing cause was the low impact resistance of the material used for the pump shaft couplings.

Applicability to Palisades:
This event describes pump failures described in the INPO EPIX search as well as the 1994 Indian Point pump failure. Elements of this event are similar to Palisades. There is currently no evidence that impact or rapid loading pump conditions were experienced at Palisades but the coupling hardness was a contributor to the September 29, 2009 coupling failure.

Non-Nuclear
An Internet search for similar non-nuclear pump failures was conducted. Similar pumps are used in water treatment plants and other facilities. Various key words were used such as "layne bowler", "vertical pump failures", and "coupling failures". The internet hits did not produce any database that discussed similar coupling failures. Relevant internet hits identified NRC documents that already have been identified above. An additional search on "410 stainless failures" did produce two articles of interest. The first article, published by Flowserve in a "Materials Newsletter", dated September 2004, discusses temper embrittlement if materials like 410 stainless steel are cooled too slowly between 800 and 1000 °F. A second article, published by the US Army Corps of Engineers, September 2003, titled "Results of Evaluation of Bolt Failures at the R.C. Byrd Locks and Dam", states:
"Type 410 stainless steel is subject to temper embrittlement during the heat treatment process. If the material is held too long in the 700 to 1,000 °F, it allows the precipitation of carbides, which reduce the toughness and increase the tensile strength and hardness. The tensile strength and hardness peak when the stainless steel is held at 885 °F for an extended period of time. If the material passes through this temperature range, then little precipitation occurs and; therefore, no embrittling effects affect the structure of the stainless steel. Even though the furnace temperature shows a fairly rapid movement through this range, the parts being heat treated can still be embrittled if they are clumped tightly together, thus affecting the overall mass that needs to be heated up or cooled down."

Applicability to Palisades
The 2009 coupling failure indicated after a review of the heat treatment facilities operating processes, it was determined that this condition does not apply to heat treated materials purchased from HydroAire and their sub-supplier Bodycote. Bodycote uses nitrogen at 2 bar (approximately 29 psi) to quench after the high heat treating step. This eliminates the possibility of the concern in the OE article. Based on pending metallurgy on the current failure, this OE may have applicability.

Additional or new Operating Experience since CR-PLP-2009-04519, "Service Water Pump P-7C Failure to Provide Discharge Pressure"
Additional INPO and Internet searches were conducted on the term "pump coupling failure." Various alterations of these terms were used to narrow the return hits to those that might be more applicable to this event. The INPO Operating Experience web site was searched using "pump failure" and the date January 1, 2009 through August 17, 2011 to assure any new events were identified.
*One additional operating experience not previously identified was found on the Internet. The "Handbook of Case Histories in Failure Analysis" identified a fracture of a coupling in a line-shaft vertical turbine pump. The pump was installed in a *dam foundation. The cause identified the fracture was brittle and initiated by an intergranular cracking mechanism. Improper heat treatment was attributed to the material being susceptible to corrosion being initiated by stress or hydrogen cracking.

Applicability to Palisades
This event involved an upper coupling with failure initiated by intergranular stress corrosion cracking of similar material. This operating experience appears to have applicability to the Palisades failure.

*The Perry Nuclear Station experienced a second service water pump coupling failure May 21, 2004. This was one of the events identified in the INPO Operating Experience Digest 2006-02 (OE used in CR-PLP-2009-04519). A copy of Perry's root cause report was obtained and reviewed for additional details.

Applicability to Palisades
This event involved an upper coupling with failure initiated by intergranular stress corrosion cracking of similar material. Design of the coupling was also investigated for stress points that may have exasperated the failure. This operating experience is applicable to the Palisades failure.

*EPIX Failure #339, Prairie Island - On 25 JUL 2010, at 10:21, the 121 Motor Driven Cooling Water Pump experienced a complete loss of pump discharge pressure due to the failure of two shaft couplings and the separation of their respective shaft segments. The first and second couplings from the pump shaft motor end were found to be fractured 360 degrees. The failure of both couplings was identified as a faulty design specification. The specification did not limit the hardness. The high hardness caused the coupling to be less tough, which subsequently reduced coupling tolerance to the affects of MIC, and increased susceptibility to transgranular and intergranular stress corrsosion cracking. Combined with MIC pitting at the relief hole, stress cracks were exposed, which caused a rapid failure of the couplings by brittle intergranular fracture

Applicability to Palisades
This event involved an upper coupling with failure initiated by intergranular stress corrosion cracking of similar material. Design of the coupling was in question. The coupling supplier is the same as Palisades. This operating experience is applicable to the Palisades failure.

Summary of Operating Experience using Grade 416 Stainless Steel

Operating experience that discussed temper embrittlement, material hardness, stress corrosion cracking and coupling design features were found to be applicable. The Palisades coupling material being 416 stainless steel is prone to these attributes and requires consideration in light of this recent coupling failure. Operating Experience examined included the Perry Repeat failures in 2003 and 2004 as well as the recent 2010 Prairie Island Event. The Prairie Island OE was used by Palisades as part of the decision to change coupling materials from 416 SS to 17-4ph SS.

The OE highlights the need to ensure that proper material specifications and processes are applied for controlling hardness, toughness and other material properties that make 416 stainless steel less prone to temper embrittlement and corrosion cracking failures.

The OE also makes it clear that Licensees stipulate proper quality controls that assure coupons and testing results reflect actual material conditions.

Also, Design Engineering activities need to verify the coupling design to assure that "stress risers" are minimal and would not contribute to corrosion cracking

<u>**Stainless Steel – Grade 630, 17-4 PH**</u>
INTERNAL and External Operating Experience:

Operating experience search for 17-4 PH material was conducted. INPO and NRC websites searches were used. Limited events with 17-4 PH steel were identified and mainly pertained to valve stems and springs in primary coolant or engineered safeguards systems environments. Examples were NRC Information Notice 2007-02, Failure of Control Rod Drive Mechanism Lead Screw Male Coupling at a Babcock and Wilcox Designed Facility, and Information Notice 86-72, Failure 17-7 PH Stainless Steel Springs in Valcor Valves due to Hydrogen Embrittlement.

IN 2007-02 event resulted from thermal embrittlement due to the component being exposed to high temperature (550°F). IN 86-72 failure of disc guide assembly springs made out of 17-7 stainless steel resulted from hydrogen embrittlement which is a function of high temperature, water chemistry, water flow condition, and time of exposure to the service condition. Other operating experience identified appeared to be similar these and not particularly relevant to the application for service water pump couplings.

OE31481 - Salem Unit 2, Microbiologically Influenced Corrosion Causes Valve Shaft Failure, Event Date: 05/17/2010
While performing the periodic component cooling heat exchanger service water side high flow flush, the service water flow could not be adjusted to the required range. Initial troubleshooting determined that the inlet flow control valve was not controlling flow as expected. The plant entered an unplanned 72 hour Limiting Condition for Operation and

replaced the valve. The causal investigation determined that the A564 Grade 630 (17-4 PH) valve shaft failed from microbiologically influenced corrosion.

Lessons Learned for the Industry:
Microbiologically influenced corrosion may affect operation of A564 Grade 630 (17-4 PH) valve shafts in brackish raw water systems.

An internet search was conducted for 17-4 stainless steel with several documents providing discussions on applications of this material and in some cases vulnerabilities of the material.

Fastener Technology International, October 2003 article, "Other Causes of Fastener Failures," presented a situation where after passivation of 17-4 stainless steel fasteners, the material appeared corroded or attacked with a dull gray surface finish. The fasteners had not been in service. A cross section etched metallographic examination revealed a distinct non-uniform white layer along the surface of the fastener that was found to be reverted austenite. Corrosion of this layer was visible. Reverted austenite is typically the result of nitrogen pick-up during heat treatment most likely from a contaminated furnace atmosphere.

NACE International, Document ID - 04126
An offshore production facility philosophy of material selection was to use corrosion resistant alloys wherever wet gas was being handled. A 17-4 stainless steel valve stem failed twelve days after startup. An operator walking by one of the main 32" ball valves heard a loud crack and saw the valve stem rise up through the top of the valve. The valve was sitting in the fully open position not being operated at the time.
The fracture was a classic sulfide stress cracking brittle failure. Insitu hardness testing of the broken stem, as well as, stems from four other similar valves on the platform found hardnesses in the range of 35 - 45 on the Rockwell C (HRC) hardness scale. Original specifications called for these stems to meet NACE MR0175, where the maximum allowed hardness is 33 HRC. Subsequent investigations found that the forging mill had taken raw 17-4 PH material, cut coupons, heat-treated the coupons and created mill specifications based on the coupons. The actual stem material was heat treated separately and incorrectly.

Stephen J. Morrow article, "When High-Strength Means No-Strength"
In this article, the author points out that for High-strength materials metallurgical factors are important to understand, but of equal importance is environmental influences which can promote failures such as environmentally induced cracking. Two major types of environmental cracking are hydrogen embrittlement (HE) and stress corrosion cracking (SCC). Both of these phenomena often result in catastrophic, brittle fracture at stress levels significantly below the materials yield stress.
High-strength steels often utilized for pump shaft applications can be susceptible to hydrogen embrittlement (HE). Hydrogen embrittlement (HE) is the general term given for a loss of toughness resulting from hydrogen absorption. Embrittlement results from the interaction of hydrogen and tensile stresses in susceptible materials. This type of hydrogen damage occurs most often in alloys such as quenched and tempered martensitic steels, and the martensitic precipitation-hardened steels. Susceptibility to fracture generally increases with increased strength and hardness. Embrittlement can result from a very small amount of hydrogen, often as little as a few parts per million.

Hydrogen may enter susceptible alloys from various sources. Some include: solutions containing hydrogen sulfide (H2S); strong acids; galvanic coupling to more active (anodic) alloys in a corrosive environment such as seawater; cathodic protection; and even microbiological (e.g. SRB - sulfate reducing bacteria) corrosion. Other sources may include residual hydrogen pick up from electroplating or pickling operations, exposure to high-pressure hydrogen gas, or sodium sulfite decomposition in high pressure boiler feed-water, etc.

Sulfur/sulfide environments not considered sour by NACE MR-0175 definitions can result in failures by hydrogen embrittlement (HE) and sulfide stress cracking (SSC) of susceptible materials such as 17-4 PH precipitation hardened stainless.

Even though materials may be selected because of compliance to NACE MR-0175, it doesn't guarantee freedom from environmental cracking. It should be noted that materials included in this standard are resistant to, but not necessarily immune to SSC under many service environments. While the susceptibility to SSC can be strongly affected by heat treatment, 17-4 PH precipitation hardened steels that have been properly heat treated to the NACE MR0175 requirements still failed by cracking. Even with NACE requirements specified there is no guarantee that failures will be prevented in 17-4 PH stainless steel. The NACE MR-0175 heat treatment requirements for UNS S17400 precipitation-hardened stainless steels requires either a Double Aging treatment at 1150°F; or a three step process which is also a Double Aging treatment at 1400°F then at 1150°F for a maximum hardness of 33 HRC. The later three step process can be furnished by specifying steel to meet ASTM A564 UNS S17400 Type 630 in the H1150M condition, rather than the single aged H1150 condition, and adding the requirement for 33 HRC maximum hardness. The resistance of high-strength steels to environmental cracking improves with reduced strength (hardness), and alloying to improve toughness. Specifying ASTM A564, Grade UNS S17400 in the H1150M condition results in significant reductions in strength; lowering the ultimate tensile strength from 135 Ksi to 115 Ksi min.; and the yield strength from 105 Ksi to 75 Ksi min.

Nickel Development Institute, "Guidelines for selection of nickel stainless steels for marine environments, natural waters and brines," states 17-4 stainless steel is widely used in marine equipment wherever higher strength is required. However, it is somewhat prone to crevice attack. Crevice corrosion is the localized breakdown of the chromium oxide film, which is caused by micro and macro biofouling organisms attaching themselves to the stainless steel surface. Overaged condition, H1100 or H11150, is preferred for better resistance to stress corrosion attack.

NUREG/CR-6223, "Review of the Proposed Materials of Construction fro the SBWR and AP600 Advanced Reactors," is mainly a review of materials being used in the primary systems of these reactor designs. It identifies uses of 17-4 PH materials in various components and the vulnerabilities mention in the previous operating experience documents. This document cautions that 17-4 PH precipitation-hardening stainless steel chosen for the control rod drive seal housing nuts in the SBWR is subject to severe SCC and hydrogen embrittlement if improperly heat treated, and stringent acceptance criteria are required for this component to avoid this potential problem.

Operating Experience Summary for using Grade 630, 17-4 PH Stainless Steel

Attachment IV – Operating Experience

The OE identified the need to evaluate coupling environment and material conditioning to assure 17-4 stainless steel coupling are not prone to hydrogen embrittlement, thermal embrittlement, sulfide stress corrosion cracking, crevice corrosion and Microbiological influenced corrosion.

Also, there is a need to assure stringent process controls are applied to coupling manufacture to avoid contaminates during the heat treat process, such as nitrogen pick-up, that would affect the conditioning of 17-4 PH material.

Additionally, the responsible design activity must ensure proper material specifications and processes are applied for controlling hardness, toughness, and other material properties that make 17-4 less prone to embrittlement and corrosion cracking failures.

Equally important is the need to stipulate proper quality controls that assure coupons and testing results reflect actual material conditions.

Attachment V: LPI Report

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Est.1885

METALLURGICAL AND FAILURE ANALYSIS OF SWS PUMP P-7C COUPLING #6

Report No. F11358-R-001
Revision DRAFT G

September, 2011

Prepared For

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Attachment V: LPI Report

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Attachment V: LPI Report

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Nondestructive Engineering

DOCUMENT RECORD

Document Type:	☐ **Calculation** ☒ **Report** ☐ **Procedure**
Document No:	F11358-R-001
Document Title:	Metallurgical and Failure Analysis of SWS Pump P-7C Coupling #6
Client:	Entergy Nuclear Operations, Inc.
Client Facility:	Palisades Nuclear Plant
Client PO No:	10325528
Quality Assurance:	**Nuclear Safety Related?** ☐ **No** ☒ **Yes**
Computer Software Used:	☐ **No**[1] ☒ **Yes**[2] — 1. Check NO when EXCEL, MathCAD and/or similar programs are used since algorithms are explicitly displayed. 2. Include Software Record for each computer program utilized
Instrument Used	☐ **No** ☒ **Yes**[3] — 3. Include Document Instrument Record

Revision	**Approval Date**	**Preparer**	**Checker**	**Design Verification**	**Approver**[4]
DRAFT G	08/31/11	S. Yim John Mills, Ph.D Ryan Chen	P. Bruck	T. Esselman, Ph.D	P. Bruck

[4] The Approver of this document attests that all project examinations, inspections, tests and analysis (as applicable) have been conducted using approved LPI Procedures and are in conformance to the contract/purchase order.

Page	2	of	xx	**Total Pages**	Include any Title Sheet and Attachments in page count

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RECORD OF REVISION

Revision No.	Date	Description of Change	Reason
DRAFT G	See Document Record	Issued for comment	

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DESIGN VERIFICATION CHECKLIST

Document No(s)[1]:	F11358-R-001				Rev.:	
Review Method:	X	*Design Review*		*Alternate Calculation*		*Test*

	Criteria	DV[2]
1	Were the inputs correctly selected and incorporated into design?	
2	Are assumptions necessary to perform the design activity adequately described and reasonable? Where necessary, are the assumptions identified for subsequent re-verifications when the detailed design activities are completed? If applicable, has an as built verification been performed and reconciled?	
3	Are the appropriate quality and quality assurance requirements specified?	
4	Are the applicable codes, standards and regulatory requirements including issue and addenda properly identified and are their requirements for design met?	
5	Have applicable construction and operating experience been considered, including operation procedures?	
6	Have the design interface requirements been satisfied?	
7	Was an appropriate design method used?	
8	Is the output reasonable compared to inputs?	
9	Are the specified parts, equipment, and processes suitable for the required application?	
10	Are the specified materials compatible with each other and the design environmental conditions to which the material will be exposed?	
11	Have adequate maintenance features and requirements been specified?	
12	Are accessibility and other design provisions adequate for performance of needed maintenance and repair?	
13	Has adequate accessibility been provided to perform the in-service inspection expected to be required during the plant life?	
14	Has the design properly considered radiation exposure to the public and plant personnel?	
15	Are the acceptance criteria incorporated in the design documents sufficient to allow verification that design requirements have been satisfactorily accomplished?	
16	Have adequate pre-operational and subsequent periodic test requirements been appropriately specified?	
17	Are adequate handling, storage, cleaning and shipping requirements specified?	
18	Are adequate identification requirements specified?	
19	Are requirements for record preparation review, approval, retention, etc., adequately specified?	
20	Has an internal design review been performed for applicable design projects? Have comments from the Internal Design Review been appropriately considered/addressed?	

(1) *Include any drawings developed from reviewed documents, or include separate checklist sheet for drawings*
(2) *Design Verifier shall initial indicating review and mark N/A where not applicable*

DV Completed By:	*Printed Name*	*Signature*	*Date*

Page ☐	**of** ☐	**Total Pages**	Include DV Checklist and Comment Resolution sheets in page count

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DOCUMENT SOFTWARE RECORD

(Include Separate Sheet for Each Software Package Utilized)

1	**Computer Software Used** (Code/Version)	ANSYS Version 11.0
2	**Software Supplier**	ANSYS, Inc.
3	**Software Update Review**	☒ Error notices; describe: Reviewed error reports for elements used ☐ Other; describe:
4	**Nuclear Safety Related Software**	☐ NO ☒ YES[1] 1. If YES: Hardware identification # used for execution: *Desktop Serial #: J2WTBM1* Basis for V & V: [16]
5	**Input Listing(s)**	☐ Input listing(s) attached: ☒ Not attached; identify File/Disc ID*: Coupling Pump Bearing & Bending.txt Coupling Pump Bearing.txt Coupling Pump No Bearing.txt *A CD with input listings and output data to be provided on project completion.
6		☐ Output results attached: ☒ Not attached; identify File/Disc ID*: *A CD with input listings and output data to be provided on project completion.
7	**Output Identifier(s)***	(see 6 above)
	*e.g., run date/time; use for reference, as appropriate, within body of calculation	
8	**Comments**	
9	**Keywords****	SOLID45, Static
	**For use in describing software features used in this calculation; use common terms based on software user manual.	
10	**Project Manager Name:**	S. Yim

If computer software was used on project, complete form with required information.
Update the LPI Computer Software Use List per LPI Procedure 13.1 requirements.

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DOCUMENT INSTRUMENT RECORD				
Instrument Used		**Instrument Description**	**Serial No.**	**Calibration Due Date**
1	☒	Tensile Testing Machine (120 kips)	Baldwin 37205	4/7/12
2	☒	Extensometer (1 in)	2620-824/1033	4/7/12
3	☒	Charpy Impact Tester	Satec Model SI-1K/1306	6/17/12
4	☒	Hardness Tester	Wilson 5YR/58	4/7/12
5	☒	Thermocouple	Omega 650 J/8320	7/12/12
6	☒	Caliper	Fowler 6"/7082002	6/21/12
7	☒	Magnetic Yoke	Magnaflux Y-6/43530	*Per use calibration*
8	☐			
9	☐			
10	☐			
11	☒	SEM/Oxford EDS	17218-118-01	*Per use calibration*
12	☐			
13	☐			
14	☐			
	Project Manager Name:	S. Yim		

For instrument(s) used on the project, identify instrument and include the instrument calibration due date.
Update the LPI *Instrument Use List* per LPI Procedure 13.1 requirements.

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INTRODUCTION/BACKGROUND

Two (2) failed coupling along with other intact components, as documented in the receipt inspection of 8/17/11 and 8/18/11 (provided in Attachment A), were submitted to **LPI** (*Lucius Pitkin, Inc.)* for material and failure assessment. The received components were extracted from Pump P-7C of the Service Water System (SWS) at Palisades Nuclear Plant (PLP). One of the two failed couplings was from a coupling failure event in September 2009 as documented in CR-PLP-2009-04519 [1][1] and the other failed coupling was from a failure event in August 2011 as documented in CR-PLP-2011-03902 [2]. The failed couplings from the 2009 and 2011 failure events are herein referred to as the "09-P7C-7F" and "11-P7C-6F", respectively (refer to coupling identification convention in Section 2.1). Photographs of couplings 09-P7C-7F and 11-P7C-6F are presented in Figure 0-1 and Figure 0-2, respectively.

The SWS comprise of three motor driven vertical multistage pumps, tagged P-7A, P-7B and P-7C, supplying water from Lake Michigan to three service water headers. All three SWS pumps are similar in design in that they are comprised of two stage stainless steel impellers coupled to the motor through six line shafts, a packing shaft and a motor shaft for a total height of over 40 feet from suction to discharge. Figure 0-3 shows the shaft and coupling arrangement for the SWS pumps and identifies couplings 09-P7C-7F and 11-P7C-6F. As can be seen in Figure 0-3, the 09-P7C-7F is coupling #7 and the 11-P7C-6F is coupling #6. A rendering that identifies the pump components (excluding the motor) is provided in Figure 0-4.

P-7A and P-7C are Layne and Bowler Model 25RKHC pumps while P-7B is a Johnston Model 25NMC pump. Each pump is driven by a 350 horsepower (HP) motor providing a rated 8000 GPM and 140 ft total developed Head (TDH) each at 50% service capacity [1].

The specified material of the 09-P7C-7F and 11-P7C-6F as well as all shaft couplings on the three SWS pumps are ASTM A582 Type 416 stainless steel (SS) [4]. The material specification for the shaft couplings on all three pumps was changed from carbon steel to 416 SS and specified with a Rockwell C hardness (HRC) value of 28 to 32 under EC-50000121762 [4] in December 2007. The couplings were also redesigned to incorporate an alignment hole that allows

[1] Numbers in brackets (e.g. [5]), indicate references listed in Section 0.

verification of proper shaft installation. The shaft couplings for P-7A were replaced on April 4, 2009 per Work Order (WO) 51637416. The shaft couplings for P-7C were replaced on June 12, 2009. The shaft couplings on P-7B were replaced during rebuild of the pump and installed in June of 2010 [2]. A detail drawing of the line shaft coupling is provided in Figure 0-5.

Scope and Purpose

The scope and purpose of this report is to provide results of the metallurgical examination and tests performed in accordance with LPI Procedure F11358-P-001 [5] and provide a probable root cause of the 2011 coupling failure. The scope of test and examinations performed are provided in the Test Matrix provided in Table 0-1.

Table 0-1: Test Matrix

Test	Test Components	
	2011 Couplings (Note 1)	2009 Failed Coupling
Visual & Photographic	1 through 7	
Surface Hardness	1 through 7	X
UT Exam	See Note 1	
Dimensional Exam	1 through 4	
Comp Analysis of Surface Deposits	6 and 7	
MT Exam	5 through 7	
Tensile Test	5 through 7	
CVN Test	5 through 7	X
Thru Thick Hardness	5 through 7	X
Comp Analysis	5 through 7	
SEM	6	X

Note(s):

1. UT examination of the couplings could not be performed due to the end geometry of the couplings. The lack of this examination in the overall Test Matrix does not diminish the capability of assessing the failure mechanism.