April 30, 2002

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

DOCKET 50-255 - LICENSE DPR-20 - PALISADES NUCLEAR PLANT
PRIMARY COOLANT PUMP CASING STUD EVALUATION – CODE CASE N-566-1

By letter dated August 26, 1999, the Nuclear Regulatory Commission (NRC) approved for Consumers Energy Company the use of American Society of Mechanical Engineers Boiler and Pressure Vessel Section XI Code Case N-566-1, "Corrective Action for Leakage Identified at Bolted Connections, Section XI, Division 1," as an alternative to the requirements of paragraph IWA-5250. This code case allows inspection and evaluation as an alternative to casing stud removal when determining suitability for continued operation with degraded bolted connections. Nuclear Management Company, LLC (NMC) is notifying the NRC that NMC is using the provisions of code case N-566-1 to support continued operation of primary coolant pump P-50C for the Palisades Plant until the next refueling outage. The prescribed evaluation was performed in accordance with subparagraph IWB-3142.4 and is submitted in accordance with subparagraph IWB-3144(b). Based on the enclosed evaluation, use of the code case for this application is acceptable.

Enclosure 1 contains the overall evaluation of the degraded bolted connections and Enclosure 2 contains the NMC engineering analysis that calculates the degradation of the bolted connections and determines acceptability for continued operation.

A047

SUMMARY OF COMMITMENTS

This letter contains one new commitment and no revisions to existing commitments.

The new commitment is:

> The primary coolant pump P-50C casing joint will be visually inspected and assessed at each plant shutdown to below Mode 4 conditions, until repairs can be completed in the next refueling outage.

Paul A. Harden
Director, Engineering

CC Regional Administrator, USNRC, Region III
Project Manager, USNRC, NRR
NRC Resident Inspector - Palisades

Enclosures

ENCLOSURE 1

NUCLEAR MANAGEMENT COMPANY
PALISADES NUCLEAR PLANT
DOCKET 50-255

April 30, 2002

Primary Coolant Pump Casing Stud Evaluation

7 pages follow

SYSTEM: Primary Coolant System

COMPONENT: Primary Coolant Pump P-50C

CLASS: ASME Class 1

INTRODUCTION:

By letter dated August 26, 1999, the Nuclear Regulatory Commission (NRC) approved for Consumers Energy Company the use of American Society Mechanical Engineers (ASME) Boiler and Pressure Vessel Section XI Code Case N-566-1, "Corrective Action for Leakage Identified at Bolted Connections Section XI, Division 1," as an alternative to the requirements of paragraph IWA-5250. This code case allows inspection and evaluation as an alternative to stud removal when determining suitability for continued operation with degraded bolted connections. Leakage in primary coolant pump (PCP) P-50B was the subject for application of the requirements of code case N-566-1 as approved on August 26, 1999, by the NRC. Nuclear Management Company, LLC (NMC) is using the provisions of code case N-566-1 to support continued operation of P-50C for the Palisades Plant until the next refueling outage. The prescribed evaluation is being performed in accordance with subparagraph IWB-3142.4 and submitted in accordance with subparagraph IWB-3144(b).

FUNCTION:

There are four PCPs (designated P-50A, B, C and D) in the primary coolant system (PCS). They are Byron-Jackson Company designed vertical, single suction, centrifugal pumps. The casing joint design and service conditions are the same for each pump. During normal operation, the four pumps circulate water through the reactor vessel that serves as both coolant and moderator for the core.

NUMBER, SERVICE AGE AND MATERIAL OF BOLTING:

Each pump is part of the PCS pressure boundary. The casing material is ASTM A 351, Grade CF8 low alloy steel. The 16 pump studs (numbered 1 through 16 sequentially around the pump casing) securing the upper casing to the lower casing are ASTM A 193, Grade B7 low alloy steel. The threaded portions of the studs are chrome plated. The stud material is considered susceptible to boric acid wastage.

CORROSIVENESS OF LEAKING MEDIUM AND ENVIRONMENTAL CONDITIONS:

The leaking medium is primary coolant. Service conditions inside the pressure boundary are 2060 psia and approximately 537 °F. PCS Boron concentration varies from approximately 1600 ppm at beginning of cycle operating conditions to potentially 0 ppm at end of cycle operating conditions. Leakage is characterized as a fine steam and water mist when the PCS is at operating conditions. The leak rate is not directly measurable. The leaking medium is characterized as moderately corrosive under these conditions.

LEAKAGE LOCATION:

Evidence of minor leakage was identified from boric acid accumulations in the vicinity of the pump P-50C casing joint prior to and during the 2001 refueling outage. Evidence of leakage was confirmed during surveillance pressure testing at the conclusion of the 2001 refueling outage (May 2001). The leakage was reconfirmed during forced outage (FO) 01-5001 in December 2001 (FO 01-5001 began in June 2001 and concluded in January 2002). The leakage at the joint was identified as the source of boric acid build-up on the PCP casing studs adjacent to the component cooling water (CCW) piping at the rear of the pump. There is no equipment other than the PCP studs and CCW piping in the immediate vicinity that is affected by boric acid accumulations.

LEAK RATE:

The total PCS unidentified leak rate determined on January 29, 2002 at 0320 hours was approximately 0.01 gpm. The leak rate at the pump P-50C casing flange contributing to this total cannot be determined. Therefore, a conservative treatment of the leakage would attribute the entire leakage from the casing flange area.

LEAKAGE HISTORY AT SIMILAR CONNECTIONS:

<u>Plant Experience</u>

A comparison of the FO 01-5001 pump P-50C leakage with historical data related to similar leakage in pump P-50A (Palisades condition report CPAL9801939) is provided in the following table and text.

Parameter	P-50C (FO-015001)	P-50A (CPAL9801939)
Leak Location	Between studs No.16 & No.1	Between studs No.16 & No.1
Stud diameter	4.575 – 4.590 inches	4.575 – 4.590 inches
Stud material	A-193, Gr. B7	A-193, Gr. B7
Stud No.16 minimum measured diameter	4.238 inches	3.720 inches
Stud No.1 minimum measured diameter	4.482 inches	3.700 inches
PCS leak rate	0.01 gpm	Approx. 0.02 gpm

A review of the table above indicates the occurrence of leakage for pump P-50C is similar to that of pump P-50A.

Industry Experience

Industry experience indicates the pump P-50C leakage will not lead to catastrophic failure. Electric Power Research Institute (EPRI) Nuclear Maintenance Applications Center (NMAC) publication TR-102748, "Boric Acid Corrosion Guidebook," April 1995, describes operating experience associated with similarly designed PCPs at the Fort Calhoun Plant. Fort Calhoun observed significant wastage on three adjacent studs on two pumps due to a leak in the same area (under a CCW pipe). The wastage on some of the studs at Fort Calhoun was such that only 20% of the stud was left remaining. This wastage was identified during a plant shutdown. The wastage on the Fort Calhoun studs was much more severe and affected more studs than is the case at Palisades. This information was also contained in NRC Information Notice, IN-80-27, "Degradation of Reactor Coolant Pump Studs."

VISUAL EVIDENCE OF CORROSION AT THE ASSEMBLED CONNECTION:

Current Physical Condition

ASME Section XI, 1989 Edition, Paragraph IWB-3517 requires stud removal when the level of wastage exceeds 5% of the cross-sectional area. The actual level of wastage is conservatively estimated to be 14% of the cross-sectional area for stud No.16 and 5% of cross-sectional area for stud No.1. These values are conservative because the minimum measured diameter was applied to the entire stud circumference. Field measurements indicate the degraded area of stud No.16 extends its entire circumference to varying degrees. The minimum measured diameter was 4.238 inches, while other areas of this stud show less wastage. The degraded area of stud No.1 measures only approximately three inches along the circumference, and a minimum measured diameter of 4.482 inches. Therefore, the actual reduction of cross-sectional area for stud No. 16 is less than 14% and for stud No.1 is much less than 5%. Stud dimensions are provided in the following table:

Pump P-50C Stud Number	Measured Nominal Diameter (inches)	Measured Minimum Diameter (inches)
No.1	4.584	4.482
No.16	4.579	4.238

Results of Visual Inspections

Pump P-50C studs were inspected during FO 01-5001. After cleaning, all studs other than No.s 15, 16, 1 and 2, were inspected for wastage and found to be in good condition. These four studs were VT-1 visually examined and measured. Only studs No.16 and No.1 were found degraded in the area just above the pump casing. Wastage did not continue into the stud hole or up toward the upper casing flange.

Additional Inspections

Additional inspection of pumps P-50A, P-50B and P-50D casing flanges and studs was performed. Stud diameter measurement and visual inspection was performed. Evidence of minor leakage was discovered at the pump P-50B casing flange in the same area of interest. However, inspection, and evaluation in accordance with IWB-3142.4 for joint integrity, indicated all pump P-50B studs were clean and no wastage was occurring.

Results of Other Test/Inspection Methods

An ultrasonic test has not been performed on the degraded studs. This method is not effective in detecting stud wastage of the magnitude affecting pump P-50C. Visual inspection and examination indicates no stud breakage in the accessible areas. Inaccessible areas are not affected by wastage due to lack of oxygen and fluid flow as documented in EPRI TR-102748, Section 3.2. Threaded areas are chrome plated, further reducing susceptibility to wastage.

TIME COMPONENTS HAVE BEEN DEGRADING:

Wastage is conservatively assumed to have begun after start-up from the cycle 14 refueling outage in December 1999, which is the last time pump P-50C is known to have been free from significant boron accumulation. Calculations are based on the 18 months from the end of the cycle 14 refueling outage to the beginning of FO 01-5001 in June of 2001. From these 18 months, three months are subtracted for forced and refueling outage time with the PCS below 210°F. Industry experience has shown that wastage rates are most severe when metal temperatures are near 212°F. A review of plant operating history records for this period of time indicates the PCS was below 210°F for approximately 59 days. An additional 30 days is subtracted to add a level of conservatism.

CHARACTERISTIC ADVERSELY AFFECTED:

Leakage is adversely affecting the stud cross-sectional area at the pump P-50C casing flange. The studs identified as No.16 and No.1 are adversely affected.

The maximum measured wastage is determined to be 0.341 inches. Using this wastage and the 15 months of wastage time, NMC engineering analysis EA-CPAL0104122-01 calculates an approximate wastage rate of 0.023 inches per month (0.272 in/yr). This value is similar to the wastage rates documented for pump P-50A. This wastage rate is reasonably consistent with those found in EPRI TR-102748, Test Reference F (page 4-23) and Test Reference K (beginning on page 4-34). A comparison of all available wastage rates associated with pumps P-50A and P-50C and available rates from the EPRI guidebook are contained in the following table:

NOTE: All wastage rates are converted to units of inches per year (in/yr) to be consistent with EPRI NMAC data.

Reference Document	Date	Stud No.1 Wastage	Stud No.16 Wastage	Data Source
EA-CPAL981067-01	05/22/98	Not estimated	Not estimated	Initial EA for P-50A.
EA-CPAL981939-01	11/19/98	0.216 in/yr	0.384 in/yr	Based on actual measurements at two dates for P-50A.
CPAL990588	05/12/99	0.120 in/yr	0.264 in/yr	Based on actual measurements when studs were retired from P-50A.
EPRI Test Ref. F	N/A	Minimum 0.108 in/yr	Maximum 0.124 in/yr	Immersion at 212°F & 4000 ppm boron.
EPRI Test Ref. F	N/A	Minimum 0.042 in/yr	Maximum 0.050 in/yr	Immersion at 352°F & 4000 ppm boron.
EPRI Test Ref. K	N/A	Minimum 0.639 in/yr	Maximum 0.833 in/yr	Directed steam spray; stud temp. < 175°F. Unknown boron.
EPRI Test Ref. K	N/A	Minimum 0.042 in/yr	Maximum 0.050 in/yr	Directed steam spray; stud temp. < 350°F. Unknown boron.
EA-CPAL0104122-01	12/29/01	0.474 in/yr	0.273 in/yr	Assumed for P-50C. Higher rate assumed for lesser-wasted stud for conservatism.

COMPONENT REPAIR SCHEDULE:

Component repair is presently scheduled for the next refueling outage.

RISK OF FAILURE:

The primary structural concern to be addressed is degradation of the pump joint integrity due to casing stud wastage resulting from the contact of boric acid on carbon steel. Stud wastage could result in increased leak rates that may exceed the limits of Technical Specification (TS) Limiting Condition of Operation 3.4.13, "PCS Operational Leakage," or radiological effluent releases greater than limits specified in TS 5.5.4, "Radiological Effluent Controls Program." NMC performed an engineering analysis to assess these studs and the integrity of the pump P-50C casing joint using field measurements. By comparing operational data with that associated with the previous pump P-50A leakage, and by reviewing inspection records, a wastage rate for pump P-50C was determined. Using this wastage rate, a linear extrapolation was made to the next refueling outage. The wastage rate is reasonably consistent with those found in EPRI TR-102748 and NMC's experience with pump P-50A. The calculation determined that joint preload was maintained at a level that ensures operational requirements are met and structural integrity is maintained.

<u>Probability of Casing Joint Failure</u>

The casing leak rate is relatively small and stable and may slowly degrade, which would be detected by PCS leak rate monitoring. Industry and NMC's experience indicates the casing joint will not fail catastrophically. Based on information provided in EPRI TR-102748, Section 8.0, and due to the location of the degraded studs (adjacent to each other) the joint will exhibit leakage prior to a catastrophic failure. Therefore, the probability of failure is very low.

<u>Consequence of Degraded Stud Failure</u>

NMC has determined from expected wastage rates that joint integrity will be maintained. The results of the finite element analysis described in EPRI Publication NP-5769, "Degradation or Failure of Bolting in Nuclear Power Plants," of similarly designed pumps constructed of similar materials and with a 16 stud casing flange similar to the Palisades' pumps, indicates that if the 2 studs in question were to fail, the adjacent (and all other) studs would remain intact. This publication also displays through the graph on page 8-31, an expected leak rate of approximately 10 gpm following failure of two adjacent studs. This leak rate is significantly less than Palisades' charging system capacity for PCS makeup (33 to 133 gpm).

Core Damage Frequency (CDF) or Large Early Release Frequency (LERF)

The risk due to the postulated failure of two adjacent studs is that of a required, controlled plant shutdown, since the expected leak rate following the failure is significantly less than Palisades' charging system capacity for PCS makeup. A conservative number of controlled plant shutdowns are incorporated in the NMC Probabilistic Safety Assessment for Palisades. The increase in CDF or LERF from operating in this condition, due to the possibility of a controlled plant shutdown, is negligible.

IMMEDIATE ACTIONS RECOMMENDATION:

Inspection Schedule

The Pump P-50C casing joint will be visually inspected and assessed at each plant shutdown to below Mode 4 conditions, until repairs can be completed in the next refueling outage.

Protective Measure to Prevent Further Wastage

No additional protective measures are recommended for pump P-50C casing studs. Protective measures, including zinc coating and shielding, applied to pump P-50A were demonstrated to be ineffective in surviving the operating environment and protecting the studs from wastage.

LONG-TERM ACTIONS:

Component repair is presently scheduled for the next refueling outage; therefore no additional actions are necessary.

CONCLUSION:

NMC concludes that the Primary Coolant Pump P-50C casing joint meets operational requirements based on the current corroded condition and using a reasonable rate of degradation.

Pump P-50C is considered operable based on visual inspection, engineering analysis of the degraded studs, industry experience with leakage at similar pump flanges, and the approved use of code case N-566-1. The analysis supporting this operability recommendation indicates pump P-50C will remain operable at least until the next refueling outage.

ENCLOSURE 2

NUCLEAR MANAGEMENT COMPANY
PALISADES NUCLEAR PLANT
DOCKET 50-255

April 30, 2002

Engineering Analysis EA-CPAL0104122-01
Evaluation of Wastage on Studs Between Casing and Cover of Pump P-50C

37 pages follow

Consumers Power
POWERING MICHIGAN'S PROGRESS

PALISADES NUCLEAR PLANT
ENGINEERING ANALYSIS COVER SHEET

EA-CPAL0104122-01

~~Total Number of Sheets~~ 16+ Attach

Title Evaluation of wastage on studs between casing and cover of Pump P-50C

INITIATION AND REVIEW

Calculation Status: Preliminary ○ Pending ○ Final ● Superseded ○

Rev #	Description	Initiated		Int Appd By	Review Method			Technically Reviewed		Rev Appd By	CPCo Appd
		By	Date		Alt Calc	Detail Review	Qual Test	By	Date		
0	Original Issue RMB 12/30/01	DM DSRIAT	12/26/01	[illegible]		✓		RB [illegible]	12/29/01		

Attachment 1- Stud Measurements, NDE Suplementary sketch etc.
Attachment 2- CPAL 0104122
Attachment 3- Stud Design Configuration, Byron Jackson dwg.
Attachment 4- Record of Telecon with Flow Serve (Byron Jackson)
Attachment 5- Fax from Flow Serve dated 12/18/98
Attachment 6- ASME III, 1965 Table N-422
Attachment 7- Palisades FSAR Table 5.2-3 (Sheet 2 of 22)
Attachment 8- Combustion Engineering - Engineering Specification #70P-005 for Primary Coolant Pumps
Attachment 9- BW/IP International Inc. to Consumers Energy, STUD ANALYSIS,12/20/98
Attachment 10- Primary coolant Pump 50C, Casing Leak Timeline to Forced Outage 01-5001
Attachment 11- EPRI, NMAC, Boric Acid Corrosion Guidelines Report TR-102748, 4/95
Attachment 12- ASME 1995 Section III Appendices, FIG. I-9.4, Design Fatigue Curves
Attachment 13- Primary Coolant Pump dwg.

DDL 1/3/02 [illegible] 1/4/02

Reference/Comment

1.0 OBJECTIVE

The objective of this EA is to evaluate the acceptability of the Primary Coolant Pump P-50C joint between the casing and the cover with local degradation of two of the 16 studs. Each stud is 4.58" diameter. CPAL0104122 documented the condition of these studs during Forced Outage 2001 and that condition report will determine operability.

2.0 APPLICABILITY

This EA is applicable to studs for any of the four (4) Primary Coolant Pumps at Palisades, provided the specific degradation and the degradation rate is enveloped by the conditions evaluated in this EA.

3.0 REFERENCES

2.1 ASME B&PV Code Section XI, 1986 Edition.
2.2 Drawings: M1-EA-5, M1-EA-5001, M1-EA-2006
2.3 Byron Jackson Tech manual, Vendor file.
2.4 ASME B&PV Code Section III, 1965 Edition.
2.5 Palisades FSAR, Rev. 23, Table 5.2-3, Sheet 2 of 22
2.6 EPRI Report (EPRI TR-104213s), Bolted Joint - Maintenance and Application Guide. December 1995
2.7 Telecon Record with Flow Serve dated 12/18/98
2.8 Fax dated 12/18/98 from Flow Serve.
2.9 ASME B&PV Code Section III, 1998 Edition
2.10 AISC Steel Construction Manual, 8th Edition
2.11 Marks' Handbook, page 5-30, Ninth ed.
2.12 EA-C-PAL-98-1939-01 Rev 0
2.13 Structural Engineering Handbook by Gaylord Jr., 1968 edition, pp 6-52 thru 6-54
2.14 NRC letter to Mr. NL Haskell ,"Evaluation of Inservice Inspection Program Releif Request No. RR -13 (TAC NO. MA4420)", dated January 28,1999

Reference/Comment

4.0 DESIGN CRITERIA

4.1 References 2.4 and 2.5 provide the design criteria.

4.2 Original detailed design calcalations for the pumps are not available.

5.0 DESIGN INPUT

5.1 During the forced outage in year 2001, the P-50C pump casing to cover joint was inspected to assess the condition of the studs and document the condition of local degradation of the two studs. Measurements show that the stud in position # 16 is 0.341 in thinner on the diameter and stud in position # 1 is thinner by 0.102". The corrosion is presumably from boric acid leaking from a degraded casing to cover gasket in the area. This adverse condition is documented by CPAL0104122. See Attach 2.

5.2 The Primary Cooling Pump P-50C is one of the four pumps that circulate water through the reactor. The joint between the casing and cover of the pump is characterized by 16, 4.58" diameter studs which reflect a measure of redundancy in bolt reactions. The studs are made of A-193, Grade B7 material with flash chrome plate and phosphate coating on the tap end. The nut material is A-194 Class 2H. The casing and driver mount/cover material is A-351 Grade CF8M and A-216 Gr WCB respectively. See Attach. 5 and Ref 2.2.

5.3 The original stud dimensions are given in Reference 2.2. Per Reference 2.3, the applied preload stress on the stud is 25,000 psi. This stress level is about 1/3 of the stud's yield strength of 75 Ksi per Reference 2.4, and stud's deformation during preload is well within the elastic range.

The stud has upset threaded ends, 4.73" in diameter per Attach 3.

Reference/Comment

5.4 The degraded stud configurations are depicted in Attachment 1 of this EA. Only two studs have visible degradation. There is no visual evidence that the casing, cover or the chrome plated threads are degraded. VT-1 was performed to determine the corroded dimensions.

5.5 In the current corroded condition, Stud # 16 measures 4.238 inches and # 1 measures 4.482 inches in minimum diameter.

5.6 Time line regarding casing leak for Pump 50C is shown on Attach 10. Wastage is assumed to begin after start- up from REFOUT 14 in 12/1999, which is the last time pump 50C is known to have been free from significant boron accumulation. Calculations are based on the 18 months from the end of the REFOUT 14 to the beginning of force outage 01-5001 in June 2001. From these 18 months, 3 months were subtracted for forced and refueling outage time with the PCS below 210 degrees F. A review of Reactor Engineering records for this period of time indicates the PCS was below 210 degrees F for approximately 59 days. An additional 30 days is subtracted to add a level of conservatism.

5.7 Design Code for the pump is ASME 1965 ed. Use of later editions of the ASME Codes is acceptable because:

A. Design philosphy has not changed.

B. The allowables are consistent.

C. Additional information has been added to enhance/clarify the code.

5.8 In the corroded condition, conservatively,the studs will experience less than 10 operational (heatup/cooldown) load cycles.

Reference/Comment

5.9 This evaluation is not proposed as a design analysis. It represents an assessment of operating data and physical component characteristics combined with design code methodology to determine component adequacy for a finite period of time i.e. one fuel cycle.

6.0 ASSUMPTIONS

Major: None

Minor:

The total wastage took place in 15 months as stipulated in para 5.6. The corrosion from now until Refout 2003 is considered to be linear with time. Because the next refueling outage is 15 months away, the corroded diameter is expected to be (4.238-0.341) 3.897 in. This assumption is similar to the the degradation rate assumed for Pump P-50A in 1998. The analysis and the "RELEIF REQUEST" was submitted to NRC. (See Ref. 2.14)

The height of the corrosion for stud # 16 is taken to be uniform over a 1.5 inch length as shown on Attach 1. Stud # 1 corroded condition is considered minor compared to # 16. Therefore, the analysis for stud # 16 envelopes stud # 1.

The other studs adjacent to the studs being evaluated have no visual signs of degradation as evidenced by the data in Attachments 1.

Reference/Comment

7.0 ANALYSIS

The ASME Codes stipulate that the bolted joint be designed for effects due to preload, operating pressure and any differential thermal expansion stresses. The code does not specify how these stresses are to be combined.

During discussions with Byron Jackson (Flow Serve, see REF 2.12) Palisades was informed that if a Finite Element Analysis model is performed for the pump casing, stud and cover to include the effects of preload and system pressure loads, the pressure loads will not add significantly to the calculated bolt (stud) stresses. Generally, the high strength bolts may be tightened to 70% of the specified tensile strength (see Ref 2.13).

The studs are over 4.5 inches in diameter, which makes them considerably stiff members of the joint. The pump is connected to massive piping and supported with very stiff structural members/components.

THERMAL EXPANSION

Because, all of the components are at or very close to the same temperature (same fluid), there is very little relative thermal expansion. Additionally, the thermal expansion loads are of secondary nature and self releiving. Therefore, these loads are neglected. (Ref. 2.9 para NB-3213.13 (1)b).

SEISMIC LOADS

Seismic inertia loads are relatively small because the massive structure will experience a very low ground acceleration for the horizontal direction and even less for the vertical direction.

Reference/Comment

DEAD WEIGHT

Dead weight of the motor affects the loads on the studs and will be considered in the analysis. The motor dead load affects the pre load of the studs.

CRITICAL DESIGN FEATURES

The critical design consideration is the preload for connections using bolts or studs. The applied preload is intended to be greater than the operating load on the studs. This ensures adequate compression of the joint and no leakage. As the bolt or stud material is removed by corrosion, the strain energy and, thus the preload force is reduced.

However, the reduction can be accepted if sufficient preload force is maintained. Preload is a displacement-dependent load. With a smaller stud cross section, the stiffness of the stud is reduced and the stress in the stud increases.

Because, the operating loads on the bolts do not always behave as theoretically designed, EPRI Report for "Bolted Joint Maint. And applications Guide" (Ref. 2.6) provides guidance to calculate a "ϕ" factor which depends on the relative stiffness of the fastener and the joint members. This "ϕ" factor is applied to the calculated pressure stress and the resultant factored stress is combined with the preload stress in the bolt, to compare with an allowable bolt stress. EPRI developed these rules to integrate the practical aspects with the theoretical design.

FUTURE WASTAGE

As stipulated in para 5.6, the wastage started in December 1999 and was noticed in June 2001, a duration of 18 months.There was approximately three (3) months downtime during that period. So, it took 15 months of operation to corrode stud # 16 by (4.579"-4.238") or 0.341". Assuming the plant will run for another 15 months until Refout 2003, the stud may corrode further.

Reference/Comment

Anticipated corrosion = 0.341"*15mo./15mo. = 0.341"

Anticipated corroded stud diameter is 4.238"-0.341" = 3.897"

This wastage rate is higher than 0.13 in/year as recommended in EPRI Report TR-102748 (Attach. 11). The higher wastage rate would cause the stud to be thinner and produce higher stud stresses. The degradation rate for this EA is similar to past rate for Pump P-50A. See NRC letter dated January 28, 1999. (Ref 2.14)

Although the second stud (# 1) has corroded less than the first, it will also be conservatively assumed to corrode to 3.897" between now and Refout 2003.

The measurement of the stud diameters taken for CPAL0104122 does not show any degradation on the other studs adjacent to these two studs. Therefore, the joint will be evaluated considering only two studs being degraded.

Figure 1 on page 9 is a conservatively postulated cross section of the degraded stud used in this EA to analyze the impact of corrosion of the two studs on the pump joint.

PRELOAD STUD STRESS

View AA on page 9 shows the stud configuration used to calculate the effective stiffness of the corroded stud with respect to the non-corroded stud.

Preload Force, F = K x Δ

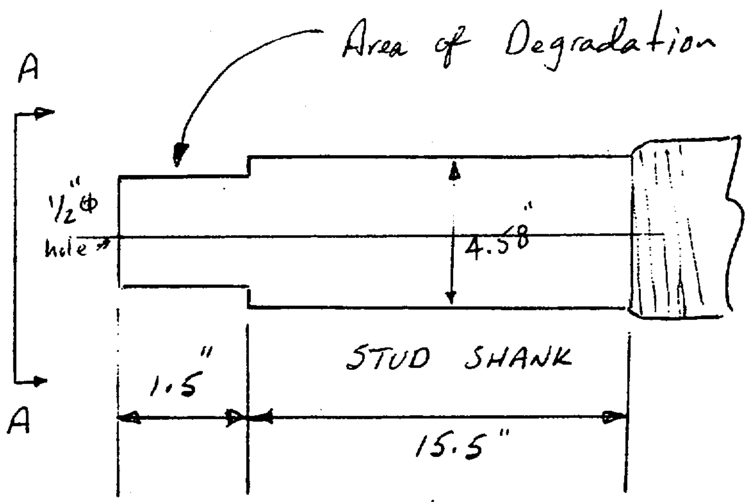
Where, Δ = stud elongation in inches
K = Stiffness = EA/L
E = modulus of elasticity
L = length
A = Cross-section Area

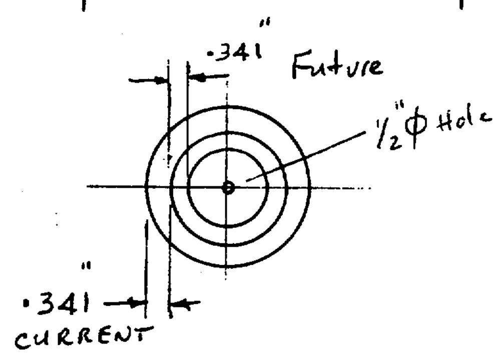
The corroded stud cross section is considered as shown on page 9. Because the wastage is uniform all around the stud, there are now two cross sections that need to be considered. Also, there is a 0.5" diameter hole in the stud. (See Attach 3)

	Reference/Comment

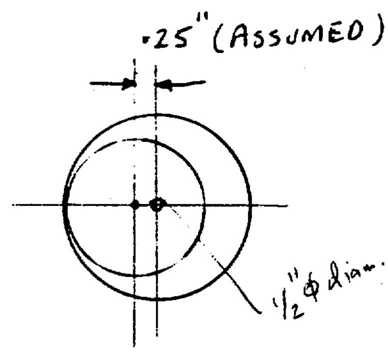
FIGURE 1

#16





SECTION A-A



FOR ECCENTRICITY

Reference/Comment

For a stepped member, the stiffness is calculated as two series connected springs. The effective stiffness Ke is given by

$$1/Ke = 1/K1 + 1/K2$$

$$K1 = EA1/L1 \text{ and } K2 = EA2/L2$$

A1 = Area of reduced cross section minus Area of 0.5" dia hole

$$= (\pi/4 * 3.897^2) - (\pi/4 * 0.5^2)$$
$$= 11.928 - 0.2 = 11.728 \text{ sq in}$$

Cross sectional area of the original stud

$$A2 = A = \pi/4\ (4.58^2 - 0.5^2) = 16.28 \text{ sq.in.}$$

$$A1/A2 = 11.728/16.28 = 0.72$$

The actual thickness of the bolted joint is 18". The 17" dimension is the length of the stud between the upset ends. (See Attach 3 and Ref 2.2)

$$L1 = 1.5L/17 \text{ and } L2 = 15.5L/17$$

$$K1 = EA1/(1.5L/17)$$

$$K2 = EA2/(15.5L/17)$$

$$1/Ke = 1/K1 + 1/K2$$

$$= (L/17E)*(1.5/A1 + 15.5/A2)$$

$$1/Ke = (L/17EA)*[1.5/0.72 + 15.5/1.0]$$

$$1/Ke = (L/17EA)*17.5833$$

$$Ke = (17/17.5833)*(EA/L) = 0.9668\ (EA/L)$$

Therefore, the effective stiffness of the corroded stud is 96.68 percent of the original stud.

Overall stiffness of the joint = $(14 \times 1 + 0.9668 \times 2)/16$
$= 0.996$

Reference/Comment

This shows that the joint effeciency with the local degradation on two studs is 99.6 percent of what it would be if there were no degradation. This shows a very insignificant decrease in the overall effectiveness of the joint and is, therefore, acceptable.

The preload stress in the corroded stud

$$= 25 \text{ X } 0.9668/0.72 = 33.57 \text{ kSI}$$

where 0.72 is the area ratio of the locally corroded stud to the non-corroded stud and 25 KSI is the initial preload.

Differential Thermal Expansion Stress

Most of the stud length is in the cover material and they both expand at the same rate. The pump casing material is different from the stud material. But, because only a small portion of the stud is inside the casing, thermal stresses, if any, are very small due to minor differential thermal expansion. Stresses due to this expansion are secondary in nature, generally self relieving and very low. Therefore, these stresses are deemed to be acceptable without further evaluation. See Ref 2.9 para NB-3213.13 (1)b.

Seismic/ Dead Weight Design Considerations

Even though, there will be some horizontal and vertical loads due to the earthquake, the loads will be very small because this pump is attached to massive and very stiff piping and supports and the combined response of the system and components reflects rigid body motion. For this reason these loads can be ignored. The external dead weight loads have been considered in the pressure stress calculations. All pertinent data is part of the Palisades FSAR Section 5.7.5.1. Following is a brief history of the NSSS seismic design:

The initial seismic design of Palisades NSSS components was conducted in a very simplistic, static manner. In the 1986 time frame, Palisades elected to use the ASME Section III,

Reference/Comment

The implementation of these curves required the determination of seismic anchor movements for piping attached to the primary coolant system. At about the same time, it was necessary for Palisades to determine time history input to the reactor vessel in order to conduct nonlinear seismic analysis on new fuel bundles. Each of these analysis demands required that the primary coolant system vessels and piping be decoupled from the stick model for the development of an interaction model. The vessels themselves (Reactor Vessel, Steam Generator, Pressurizer and Primary Coolant Pumps) were characterized by stick model members. This enabled the analyst to calculate seismic loads directly at critical cross sections.

This experience demonstrated that the interaction of the primary system components with the concrete internal structure was such that the combined system/structure (original lumped mass) model was adequate for overall structural response and seismic loading as previously calculated by the static ZPA methods was acceptable. The seismic loading was very modest and noncontrolling with respect to the other load cases and the seismic anchor movements of the primary coolant components could be ignored. This is the basis for the conclusion that seismic loadings cases and combinations are not limiting or significant in the analysis of primary system components.

Pressure Stress

Operating pressure = 2060 psi

Operating pressure force = $(\pi/4)*48^2*2060$ = 3,727.7 Kips

where the effective pump diameter for pressure is 48 inches. (See Attach 4)
Load of equipment on the joint = 122.3 Kips

Net pressure force = (3727.7-122.3) equals 3605 Kips

Force per stud = 3605/16 = 225.3 Kips

Tension stress in the corroded stud = 225.3/11.728 = 19.21 Ksi where 11.728 is the cross sectional area of the corroded stud in square inches (see Preload stud stress calculation).

Reference/Comment

Allowable stress intensity, Sm at 600 deg F = 19.8 Ksi (Attach 6)

Pressure stress is below the ASME 1965 Code Sm.

Stress Combinations

Operating Pressure + Preload

For design pressure or operating pressure conditions, the pressure stresses and differential thermal stresses are not directly additive to the preload stress.

Attachment 3 to this EA shows the original design details of the stud.

Reference 2.6 provides guidance to calculate the factor by which the pressure stresses should be modified.

Refer to page 6-21 of Reference 2.6

Le = grip length + 1/2 engaged thread length + 1/2 bolt head thickness.

$Le = 18 + 10/2 + 4.58/2 = 25.29"$

4.58" is the assumed thickness of the nut.

$Le/D = 25.29/4.58 = 5.52$

From Fig 6-17, page 6-23 of Reference 2.6, for Le/D equal to 5.52, the ratio Kj/Kb is 3.

From page 6-21, equation 6-4 of Reference 2.6,

$\phi = Kb/(Kj + Kb)$

$1/\phi = Kj/Kb + 1$

$= Kj/Kb + 1$

$= 3 + 1$

Reference/Comment

= 4

$\phi = 0.25$

The preload stress plus pressure stress in the locally corroded stud = 33.57 + 0.25 X 19.21 = 38.37 Ksi

Reference 2.4 specifies the allowable stress at the operating temperature equal to 39.6 Ksi which is two times Sm at 600 degrees operating temperature.

The calculated stress in the corroded stud using the operating pressure and preload is 38.37 Ksi which is less than the allowable stress intensity. This is acceptable and, therefore, the corroded stud can take the stress imposed on it by the preload and operating pressure.

<u>Bolt Eccentricity</u>

For stud # 16, there is no eccentricity because the wastage is 360 degrees around the stud in a uniform fashion.The fact that the wastage is only on one side of the stud #1 may cause an eccentric load path through the corroded length of the stud. But the wasted area for that stud is much smaller and the height and the width of the wastage is also less than one-half of stud # 16. So, the insignificant eccentricity may cause small local bending in the stud, but the resulting moments will be minimal and do not impact the load carrying capacity of the studs as shown below. The two adjacent degraded studs being less stiff than the other 14 may cause a slight joint asymmetry. But this is too minute to be a concern.

Assume that there is a 0.25 in eccentricity from the center for the bolt load path. This will provide additional design conservatism for stud # 16.

NOTE: Similar evaluation was performed for Pump P-50A in 1998 where there was eccentricity. P-50A analysis was submitted to NRC in 12/98. See Ref. 2.14

Reference/Comment

NOTE: Similar evaluation was performed for Pump P-50A in 1998 where there was eccentricity. P-50A analysis was submitted to NRC in 12/98. See Ref. 2.14

Max. Bending moment M due to eccentric path
= 38.37* 11.728* 0.25
= 112.50 in.kip

Moment of inertia, $I = \pi/64(3.897^4 - 0.50^4) = 11.3181\ in^4$

Section Modulus, $S = \pi*(3.897^4 - 0.50^4) / (32*3.897)$
$= 5.8086\ in^3$

Bending stress due to moment = (112.50) /(5.8086) KSI
= 19.368 Ksi

Stud Cyclic (FATIGUE) Evaluation

NB 3222.4 (e) of Reference 2.9 states that a strength reduction factor for fatigue need not be greater than 5. So by using a factor of 5,
Peak Bending Stress = 19.368* 5 = 96.84 ksi

Total Peak Stress = 38.37 + 96.84 = 135.21 KSI

Peak Alternating Stress = 135.21/2 = 67.60 KSI

FIG. I-9.4 of Reference 2.9 shows two design fatigue curves for high strength steel bolting for temperatures not exceeding 700 degrees F. One curve relates alternating stress values and number of cycles for maximum nominal stress less than or equal to 2.7 Sm. The other curve relates alternating stress values and number of cycles for maximum nominal stress equal to 3.0 Sm. Using the conservative curve the number of allowed cycles is more than 1000 cycles. See Attachment 12.

Number of cycles, experienced by the studs in the corroded condition is less than 10. Therefore, usage factor = 10/1000 = 0.01 which is extremely small as compared to 0.1 allowed in the industry practice.

Reference/Comment

8.0 CONCLUSION

Based on the current corroded condition and using a linear rate of degradation until REFOUT 2003, it is concluded that the Primary Coolant Pump P-50C joint meets the ASME Section III allowables and, therefore, complies with the FSAR requirements.

This EA has made some conservative assumptions fom the analytical standpoint such as the number of cycles for the studs and assumed stud eccentricity. Based on this analysis, it is concluded that sufficient preload still exists (and will exist until such time when the studs are replaced i.e. 3/2003) to prevent the joint from separating under operating conditions.

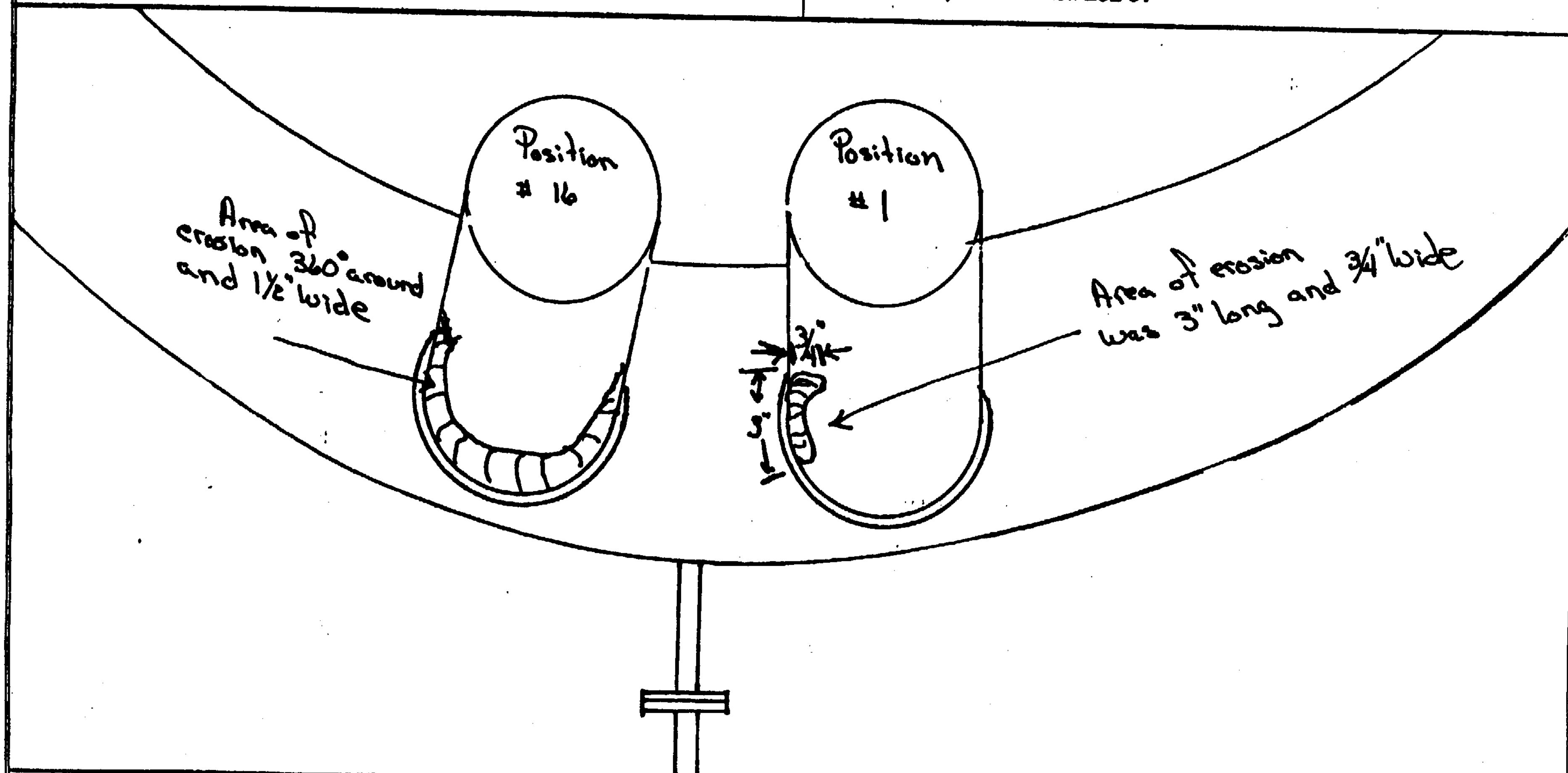
Attach 1 sh 1 of 5
EA-CPAL0104122-01

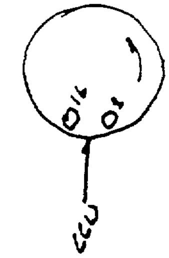
Consumers Energy

Laboratory Services

NDE Supplementary Sketch

Examiner: *Brian Lenius*	Level: *II*	WO/MO No.: *n/a*	Date: 12-27-01
Project No.: *0100510*	Item Type No.: *Pump Studs*	Attachment Sheet No.: *BJL-01*	
Line No.: *P-50C*		Examination Report Sheet No.: *BJL-01*	





EXAMINATION OF PCP CASING FLANGE BOLTS

CPAL0104122

(These instructions are provided for guidance. Field personnel may add information as conditions may require.)

For P-50C, perform Visual Examination VT-1 and record thinnest dimensions for bolts 15, 16, 1 and 2 as shown in Permanent Maintenance Procedure PCS-M-47, Page 71.

PUMP	VT-1 Complete Indicate by a ✓	Micrometer Reading (inches) Nom.	Min.
P-50C	15 -	15 - 4.583	4.581
	16 -	16 - 4.579	4.238
	1 -	1 - 4.584	4.482
	2 -	2 - 4.581	4.579

Visually scan the remainder of the casing flange bolting for boric acid accumulations. **IF** boric acid is present, **AND** the location is accessible, clean, perform Visual Examination VT-1 and obtain micrometer readings per the following table. Otherwise indicate **Not Required.**

PUMP	VISUAL SCAN, BA PRESENT? Yes, No or Not Accessible	CLEANED ? Yes or Not Required or Not Accessible	VT-1 COMPLETE? Yes or Not Required Not Accessible	MICROMETER READING (Bolt # and inches) Nom.	Min.
P-50C	3 -	3 -	3 -		
	4 -	4 -	4 -		
	5 -	5 -	5 -		
	6 -	6 -	6 -		
	7 -	7 -	7 -		
	8 -	8 -	8 -		
	9 -	9 -	9 -		
	10 -	10 -	10 -		
	11 -	11 -	11 -		
	12 -	12 -	12 -		
	13 -	13 -	13 -		
	14 -	14 -	14 -		

Attach #1 sh 3 of 5
EA-CPAL0104122-01

Consumers Energy
Laboratory Services

A2LA ACCREDITED

NDE –Visual Examination Report

Examiner: *Brian Lenius*	Level: *II*	Receipt Date: *8-29-01*	Exam Date: *12-24-01*	Report Issue Date:	Sheet No.: *BJL-01*
Examiner: *n/a*	Level: *n/a*	NDE Company: *Consumers Energy*		Total Hours Worked: *n/a*	
Project No. : *0100510*	Customer Location/Address: *Palisades Nuclear Plant 27780 Blue Star Mem. Hwy. Covert, MI. 49043*			Examination Location: ☐ Laboratory ☒ Customer Facility	
NDE Procedure: *NDT-VT-01*	Rev: *14*	Method: ☒ Direct ☐ Remote		Technique: ☒ VT-1 ☐ VT-3 ☐ Other:	
		Surface Condition: *as found*			

Evaluation Requirements				Light Meter: *005931*	Cal. Due: *5-25-02*	Intensity at Surf: *200 f/c*
Code: *ASME*	Year: *1989*	Section: *XI*	Part: *IWB-3517*	Observation Distance: *12" to 24"*	Observation Angle: *45 to 90*	Procedure Adequacy Demonstrated: ☒ Yes ☐ No

Other Reference: *n/a*

Material Type: *CS* | Joint Design: *n/a*

Nominal Diameter: *4.75"* | Nominal Thickness: *33"*

Item Type: *Pump Stud*

Visual Aids

- Flashlight ☒
- Mirror ☒
- Ruler ☒
- Pit Gauge ☐
- Fillet Gauge ☐
- Contour Gauge ☐
- Binoculars ☐

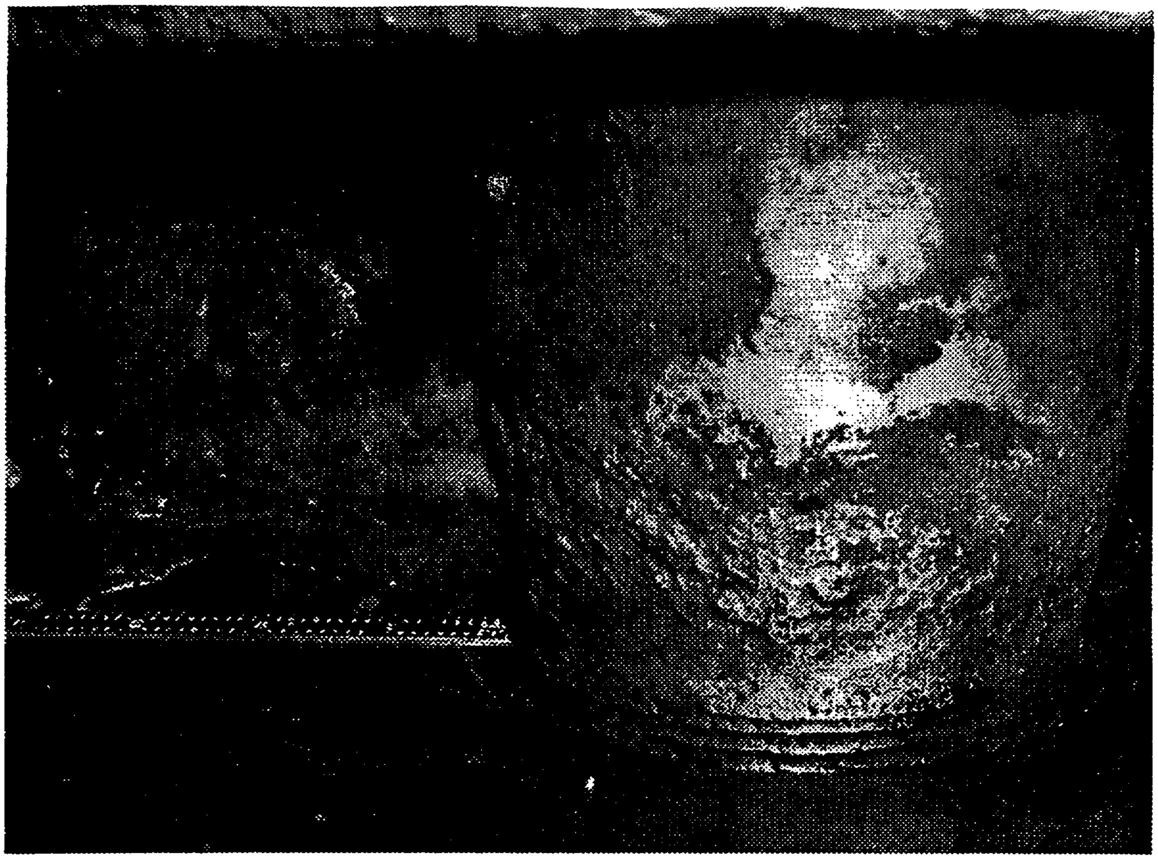
Examination Checklist

☒ Scatches/Cuts/Gouges	☒ Connection Integrity	☐ Weld Profile
☐ Lack of Fusion	☐ Weld Reinforcement	☒ Wear
☒ Linear Indication	☒ Physical Damage	☐ Undercut
☒ Physical Displacement	☐ Clearance Verification	☒ Cracks
☒ Loose/Missing Parts	☒ Erosion	☒ Corrosion
☐ Debris	Other Visual: *Other Abnormalities*	

Item ID Number	System Name	Line No.	Indication No.	Lo. Location	Indication Type	Indication Size	Weld Leg Lengths		Evaluation Acceptable	Lo. Rule No.	WO/MO No. *n/a* Remarks
1	PCP	P50C	1	8"	erosion	3"	n/a	n/a	No	4	Evidence of leakeage was noted around studs 16 and 1
2	PCP	P50C	n/a	n/a	n/a	n/a	n/a	n/a	Yes	4	
15	PCP	P50C	n/a	n/a	n/a	n/a	n/a	n/a	Yes	4	
16	PCP	P50C	1	0"	erosion	15"	n/a	n/a	No	4	See attachment BJL-01 for sketch.

A2LA Accreditation Certificate Number 1097.03 for ANSI B31.1, AWS D1.1, AWS D1.3, and ASME Section V, Article 9.

Examiner: Brian Lenius	Page 1 of 1	Reviewed By: Pending Level III Review	Level:	Date:



Stud #16 taken on December 22, 2001

Attach #1 sh 4 of 5
EA-CPAL0104122-01

Stud #1 taken on December 22, 2001

Attach #1 sh 5 of 5
EA-CPAL 01041 22-01

Attach # 2 sh 1 of 2
EA-CPAL 0104122-01

Condition Report

CPAL0104122

Title:
BORIC ACID ACCUMULATION NEAR PRIMARY COOLANT PUMP P-50C COVER CCW FLANGE

Discovery Date and Time: 2001-12-21 08:00 **Condition Discovered By:** HEALTH PHYSICS

System(s) Affected: CCS PCS

Component(s) Affected: P-50C

Description of Occurrence or Condition:
On December 21 "A" Shift, Health Physics personnel observed an accumulation of material below the Primary Coolant Pump P-50C cover Component Cooling Water (CCW) connection flange. This is located at approximately 618' elevation, behind the pump mechanical seal, outside and below the motor driver mount. Until very recently, the area was only accessible by traversing slippery pump insulation covers, or by passing directly by the pump shaft when the pump was tagged out (now an expanded metal grating allows safer access). The only known potential sources of such an accumulation in that area are boric acid from the PCS or nitrites from CCW. There were also whitish coatings on smaller piping above the accumulated buildup, a piece of rag, and possible wire debris.

Immediate Action Taken:
The observer took digital photographs of the area and reported observations after exiting. The System Engineer was notified upon arrival for day shift. A planner with primary coolant pump engineering and maintenance experience inspected the area later that morning. The System Engineer, planner, ISI lead, Health Physics, and a System Engineering section head met that afternoon to plan cleanup, fastener inspection, and leak inspection. The team, which had extensive experience with primary coolant pumps, agreed that further inspection was necessary for fastener evaluation and possible leakage, but the accumulation probably was old boric acid that had built up from many years of pump seal leaks and drips from seal instrument lines. A System Engineer jump was planned for "A" Shift December 22 after PCS pressurization to 250 psia.

System Engineer Walkdown Report: Buildup appeared to be old boric acid. Deconners had cleaned the pump flange so stud-cover interfaces were visible in the vicinity of the CCW line. There was moisture around one stud, but it appeared to result from cleaning, not a leak. No flow was noticed. A deconner stated afterward that considerable liquid had been used to try to clean off the area. One stud had what appeared to be relatively shallow surface corrosion, and there was some pump cover wastage around that stud hole. The lower two of four CCW studs were still partly buried in the boric acid "rock" and could not be more closely examined. There was no evidence of CCW leakage; the pressurized pipe and gasket area were dry. The P-50D CCW flange area was also examined since it is the only pump that has not been opened since initial criticality. There was no boric acid buildup right at the CCW flange, but there was considerable old boric acid buildup a few inches away beneath insulation covers. Loosened boric acid had fallen to the 607' area below the pump. P-50D studs did not appear degraded. There was some maintenance debris in the area. Health Physics was notified.

Recommendations (Operability and corrective Action):
History:
P-50C internals were replaced in 1985. Since then, the pump cover and CCW flange have not been disassembled. Some dry buildup was visible from the 625' floor near P-50C at the beginning of the present outage, but it did not appear to be any different from what has been seen before. A close examination of the flange area was not made at that time for safety reasons.

Attach #2 Sh 2 of 2

EA-CPAL 0104122-01

All four primary coolant pumps have tended to accumulate boric acid in this area in the past. A close examination of all primary coolant pump cover flanges in 1998 did not show any evidence of cover leakage except on P-50A (ref. CPAL9801939).

Recommendations:
1. (Decon Crew) Remove more boric acid to fully expose the two lower CCW studs and fasteners for examination.

2. (Engineering Programs) Evaluate CCW fasteners, rusted pump cover stud, and stud hole for wastage & acceptability.

3. (Operations) If Engineering determines fasteners to be acceptable, consider P-50C OPERABLE.

4. (Engineering) After final cleanup, take more pictures. Compare to 2003 REFOUT pictures to determine if boric acid buildup is continuing.

References:
CPAL980000968, CPAL9801067, CPAL9801080, CPAL9801939

Problem Resolved ☐ Yes ☐ No

Initiator: BEMIS DA **Origination Date:** 2001-12-22 02:50

Does the condition involve an equipment or programmatic issue related to the ability of an SSC to perform its safety or safety support function?

☐ Yes - Complete Admin 4.13 Attachment 1 ☐ No

Immediately Reportable?

☐ Yes - Complete Admin 3.03 Attachment 5 ☐ No

Safety Assessment per Maintenance Rule Policy Required? ☐ Yes ☐ No

Reportable: ☐ Yes ☐ No **10CFR Part#** ____________

PRC: ☐ Yes ☐ No **Licensing** ________/______

Maintenance Rule Applicable? ☐ Yes ☐ No

Significance Level: (Circle one) 1 2 3 4

Industry Experience? ☐ Yes ☐ No

Does past operability need assessment? ☐ Yes ☐ No If yes, CRTL will ensure completion

Comments: N/A

CRG Chair: ______________________________ **Date:** ____________

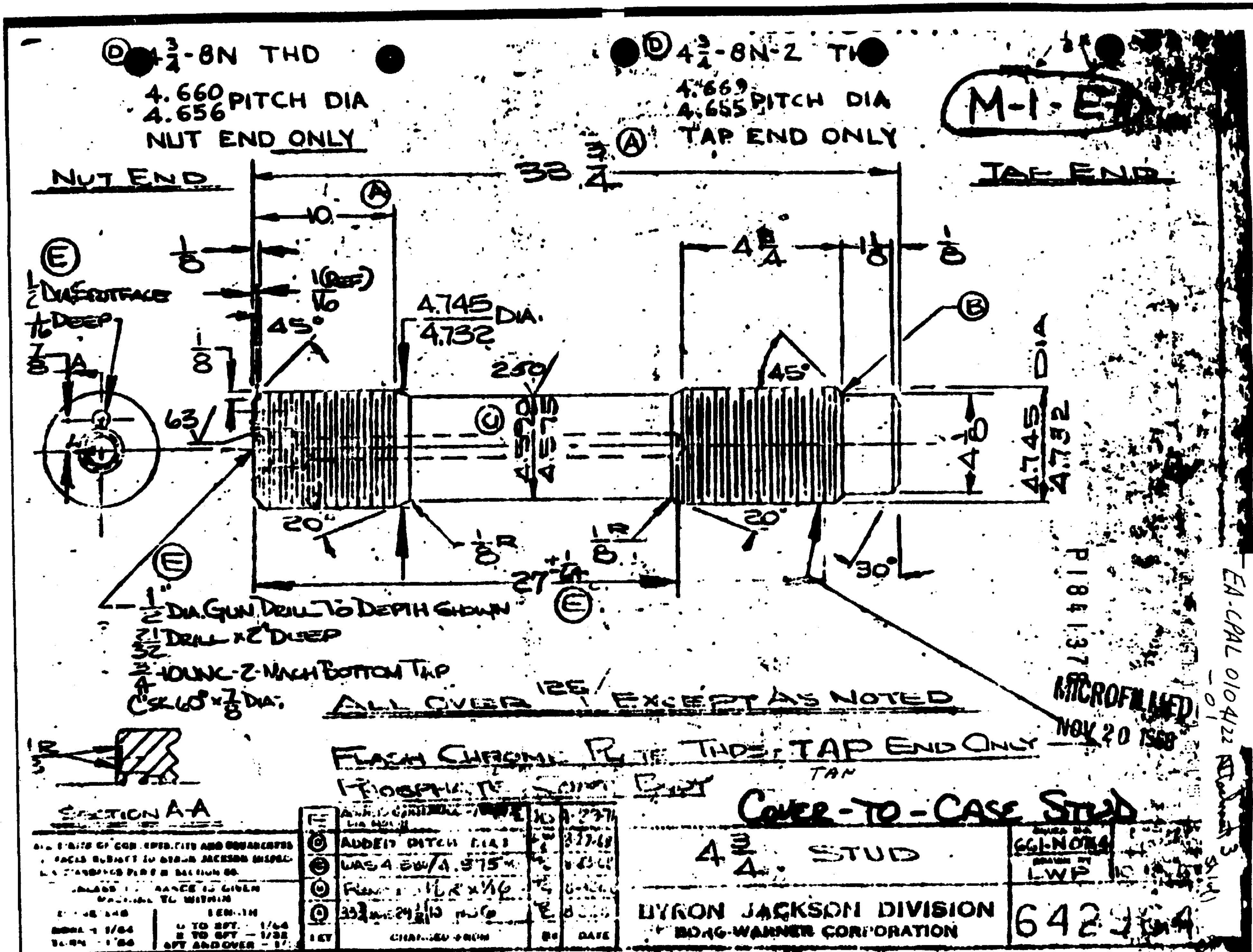
Assigned to: ______________________ **Due Date:** ____________

Evaluated by: ______________________ **Date:** ____________

Approved by: ______________________________ **Date:** ____________

CARB Chair Approval: ______________________ **Date:** ____________

Closeout by: ______________________ **Date:** ____________



EA-CPAL 0104122-01

[redacted]

E-CPAL0104122-01

Telephone Conversation with Flowserve, December 18, 1998

Attachment 4
Sh. 1 of 1

SUMMARY

On December 18, 1998, Palisades conducted a telecon with engineering personnel from Flowserve Corporation to discuss the draft review of EA-C-PAL-98-1939-01, "Evaluation of Corrosion on Studs between Casing and Cover of Pump P-50A, Reference C-PAL-98-1939." Flowserve participants included Frank Costanzo, Manager of Engineering, Pump Division; and Gerard Lenzen, Senior Project Engineer, Pump Division. Among the Palisades participants were Don Riat, Al Lyon, Raj Gupta, and Don Bemis. Other people came and went during the conversation.

DISCUSSION

Flowserve comments on the draft calculation were discussed. In general, the comments were similar to comments generated internally at Palisades.

Flowserve assumes the hydraulic lifting area to extend to the outer diameter of the outer casing gasket, which is 48 inches. This is based upon the conservative assumption that the inner gasket is nonfunctional, allowing full pressurization up to the outer gasket. They believed that a slightly smaller effective diameter would be justifiable in accordance with standard gasket principles.

The code maximum yield stress allowable for the stud at the assumed temperature (600° F.) is 61 ksi.

Flowserve engineers said that we should not add bolt preload stress to actual pressure stress. Finite element analysis shows only a marginal increase in bolt stress due to pressure. The larger of the two stresses (pressure or preload) should govern design.

Section 2B of the original code stress report dealt with bolting design. Unfortunately, Flowserve cannot physically locate that section.

Based upon Flowserve's experience with pump maintenance, preload relaxation is very hard to quantify. Their suggested value is 15%. This is based upon tensioning the stud in order to determine the lift-off load.

Stud temperature would be close to pump operating temperature inside the flange. In the exposed area, it may be a hundred degrees cooler.Most plants insulate these studs and flanges, but those at Palisades are uninsulated. Due to the nature of the localized boiling, surface temperature of our stud is cooler in the exposed area.

Flowserve also provided a fax detailing pump and stud materials, as well as gasket dimensions.

E-CPAL0104122-01
Attachment 5
Sh 1/1

FLOWSERVE CORPORATION

Byron Jackson® Pumps United Centrifugal Pumps

FAX TRANSMITTAL

Date 18 Dec 98 Pages 1

To Dan Bemis
Palisades Nuclear Plant
616 764-2231

From GERRY LENZEN
Technical Services Group
Phone 213-584-1879
Fax 213-589-2529

Subject: Palisades RCP

The following is some information requested on the Palisades RCP.

Driver mount material - A-216 Grade WCB
Cover material - A-351 Grade CF8M
Case material - A-351 Grade CF8M
Stud material - A-193 Grade B7
Nut material - A-194 Class 2H

Outer gasket groove - 46" ID and 48-1/8" OD
Outer gasket - 46-1/16" ID and 48-1/16" OD
Inner gasket groove - 43-1/2" ID and 45" OD
Inner gasket - 43-9/16" ID and 44-15/16" OD

The thickness of the gasket is 0.175".
The depth of the gasket groove is 0.125".

ASME III - 1965

EA-CPAL0104122-01

Attachment 6

Sh 1 of 2

TABLE N-422

DESIGN STRESS INTENSITY VALUES, S_m, FOR STEEL BOLTING MATERIALS*

For metal temperature not exceeding °F									
100	200	300	400	500	600	700	800	900	1000
[illegible]	33,000	31,400	30,300	28,800	27,700	26,300	24,400	...	...
[illegible]	29,800	28,400	27,400	26,200	25,100	23,800	22,100	...	...
[illegible]	23,500	22,400	21,600	20,700	19,800	18,800	17,400	...	...
[illegible]	34,000	33,200	32,500	31,800	30,900	29,400	27,700	...	...
[illegible]	30,700	30,000	29,400	[illegible]	27,900	26,600	25,100	...	...
[illegible]	27,500	26,900	26,300	[illegible]	25,000	23,800	22,400	...	...
[illegible]	33,000	31,900	30,600	29,500	28,100	26,400	24,200	...	...

TABLE N-423

(In course of preparation)

EA-CPAL 0104122-01

TABLE N-422
DESIGN STRESS INTENSITY VALUES, S_m, FOR STEEL BOLTING MATERIALS*

Spec Number	Grade	Nominal Composition	Diameter In.	Min Tempering Temp °F	Min Tensile Str PSI	Min Yield Str PSI
Low Alloy Steel						
SA-193	B7	1 Cr-0.2 Mo	2½ and under	1100	125,000	105,000
			Over 2½ to 4 inc.	1100	115,000	95,000
			Over 4 to 7 inc.	1100	100,000	75,000
			2½ and under	1200	125,000	105,000
SA-193	B14	1 Cr-0.3 Mo-V	Over 2½ to 4 inc.	1200	110,000	95,000
SA-193	B16	1 Cr-½ Mo-V	Over 4 to 7 inc.	1200	100,000	85,000
SA-320	L43	2 Ni-0.8 Cr-¼ Mo	4 and Under	...	125,000	105,000

*The allowable stress values for bolting materials given in this table do not exceed the lesser of one-third of the specified minimum yield strength or one-third of the yield strength at temperature, with credit granted for the enhancement of properties produced by heat treatment. They are intended for use in the design formulas of Appendix II. For allowable values of actual preload and service stresses, see N-416.

TABLE 5.2-3
(Sheet 1 of 18)

MECHANICAL
SYSTEM/COMPONENT CLASSIFICATION

System/Component	Seismic Class per RG 1.29 Interpretation(a)	CP Co Design Class(b)	Interpretation(c)	Class per RG 1.26 Standards Used in Plant Design
REACTOR COOLANT SYSTEM				
Reactor Vessel	Category I	Class 1	ASME III Class 1	ASME III (1965) - Class A
Reactor Vessel Supports	Category I	Class 1	ASME III Subsection NF	-
Steam Generators - Tube Side	Category I	Class 1	ASME III Class 1	ASME III (1965) - Class A
Steam Generators - Shell Side	Category I	Class 1	ASME III Class 2	ASME III (1965) - Class A
Steam Generator Supports	Category I	Class 1	ASME III Subsection NF	-
Pressurizer	Category I	Class 1	ASME III Class 1	ASME III (1965) - Class A ASA B31.1 (1955)
Primary Coolant Pumps (PCP)	Category I	Class 1	ASME III Class 1	ASME III (1965) - Class A Standards of Hydraulic Institute (SHI) ASA B31.1 (1955) ASA B16.5 (1961)

(a) Seismic category as identified in the Franklin Research Center Technical Evaluation Report TER-C5257-428 Pursuant to SEP Topic III-1 and other related materials.
(b) [illegible] APPENDIX A, and TE-C5257-428.
(c) Class pursuant to the [illegible] Code, Section III, Division 1, Subsection [illegible] 1977 edition, 1978 addenda, as determined by TER-C5257-428, pursuant to SEP Topic III-1 and modified by CP Co.
(d) Current design requirements for non-PCS piping are reconciled to ANSI B 31.1, 1973 Ed with 1973 Summer Addenda. See Section 5.10.1 and 5.10.2.
(e) [illegible] Stations 1 and 2 [illegible] a seismic design from an earlier revision of the FSAR not consistent with the requirements in Section 5.2.2 of the current revision of the FSAR. The design of Stations 1 and 2 [illegible] used 0.5 x [illegible] for seismic design.
(f) Seismic Class I [illegible] on Engineered Safe-guards Systems.

Palisades FSAR

Attachment 1
Sh 1/1

EA-CPAL 010412-01

[redacted]

PCS REF 45 CQH
PART I M1EA

EA-CPAL0104122-01
Attachment 8
Sh 1 of 3

Ref 45

ENGINEERING SPECIFICATION FOR A PRIMARY COOLANT PUMP FOR CONSUMERS POWER COMPANY

CONTRACT - 2966

COMBUSTION ENGINEERING, INC.
WINDSOR, CONNECTICUT

Prepared By	/s/ E. P. Flynn	Date 7/15/66
Approved By	/s/ J. D. Crawford	Date 7/15/66
Approved By	/s/ W. K. Wilhelm	Date 7/15/66
Certified By	/s/ T. L. Katting	Registration No. 029729

State of New York

	Revision 1	Revision 2	Revision 3
Prepared By	/s/ V. S. Kreutzig	[illegible]	
Project Manager		JC Hale	
Approved By	/s/ W. K. [illegible] 1/30/67	[illegible] 4/26/68	

				BY	CK	
25	3	1-9-85	REV. PARAGRAPH 5.4 PAGE 15 of 31	DAH	JQD	[illegible]

Specification No. [illegible]

Revision 3

950Y186 * M1EA

Pg 1 of 35

[redacted]

0950 EA-CPAL 0104122-01
Attachment 8
Sh 2/3

3.0 **REFERENCE**

3.1 References not attached.

3.1.1 ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels plus addenda.

3.1.2 ASME Boiler and Pressure Vessel Code, Section II, Material Specifications plus addenda.

3.1.3 ASME Boiler and Pressure Vessel Code, Section IX, Welding Qualifications plus addenda.

3.1.4 Standards of the Hydraulic Institute - Centrifugal Pump Section.

3.1.5 NEMA, ASA, and IEEE Standards.

3.1.6 ASA-B 16.5 Steel Pipe Flange and Flanged Fittings.

3.1.7 ASA-B 31.1 Code for Pressure Piping.

F539 1147

EA-CPAL0104122-01
Attachment 8
Sh 3 of 3

4.0 DESIGN REQUIREMENTS

4.1 Equipment Requirements

4.1.1 The pump shall be a vertical centrifugal type with bottom suction and horizontal discharge. The driving motor shall be of the conventional induction type. Provisions shall be made for the collection and removal of shaft seal leakage. Suitable self-contained lubrication system for pump and motor must be included with the pump assembly. The pump and its performance characteristics shall be suitable for parallel operation with identical units.

4.1.2 The design life of the pump shall be forty years. The predicted life of those positions of the pump assembly with less than a forty year life shall be listed in the technical manual.

4.1.3 The pressure containment design of the pump assembly shall conform to Section III of the ASME Boiler and Pressure Vessel Code.

The pump casing, pressure housing, and suction elbow shall be considered as a Class "A" vessel per paragraph N-131 of Section III. 1

4.1.4 The pump shall be designed to require maintenance at a minimum interval of one (1) year. It shall be a design objective to require maintenance at three year intervals.

4.1.5 Provisions shall be made to change seals without draining the pump casing. The pressure inside the casing will be between atmospheric and 20 psig.

4.1.6 The pump assembly shall be designed for continuous operation at any point on the characteristic curve from 80,000 to 110,000 gpm including operation at specific gravity 1.0 as specified in 4.1.7.

4.1.7 The pump assembly shall be designed to operate for 5,000 hours over its design life with a primary coolant specific gravity of 1.0. Continuous operation for 250 hours under these conditions shall be within the design capabilities of the pump assembly.

4.1.8 Pump assembly shall be designed for counter-clockwise rotation when viewed from above.

EA-CPAL 0104122-01

0950 4 Attachment 9
Sh. 1 of 1

BW/IP International, Inc.

Pump Division

FACSIMILE TRANSMISSION COVER SHEET
BW/IP INTERNATIONAL, INC
BYRON JACKSON PUMPS
NUCLEAR PRODUCTS OPERATION (NPO)
2300 E. VERNON AVE.
VERNON, CALIFORNIA 90058
PHONE # : (213) 587-6171 -- FAX # : (213) 589-2080

FAX NO.: 616-764-2251 DATE: 20 December 1998
TO: Paul Harden FROM: Frank J. Costanzo
COMPANY: Consumers Energy - Palisades Nuclear Plant
SUBJECT: Stud Analysis
REFERENCE:

This is to document that we have completed our technical review of Consumers Energy report EA-C-PAL-98-1939-01 entitled "Evaluation of Corrosion on studs between casing and cover of Pump P-50A" and concur and approve this report based upon the incorporation of minor comments previously discussed and agreed upon.

Best Regards,

Frank J. Costanzo
Manager of Engineering
Nuclear Products Operations

CC: K. Lemmon
G. Lenzen
C. Reimers

PAGES TO FOLLOW: 0

DEC 20 '98 13:05 1 714 9687572 PAGE.001

Attach # 10

EA-CPAL 0104122-01

PRIMARY COOLANT PUMP 50C CASING LEAK
TIMELINE TO FORCED OUTAGE 01-5001

Refueling Outage #13; June 1998

Refueling outage pressure test RT-71A did not record any indication of leakage or evidence of leakage in the area of interest for pump 50C.

Primary Walk-Down, December 26, 1998

System Engineering Mode 3 walk-down of P-50C indicated "No sign of leakage or boric acid buildup in the flange area corresponding to the P-50A leak.

Primary Walk-Down, May 8, 1999

System Engineering Mode 3 walk-down records indicate P-50A and P-50B were closely examined in the area near the CCW pipe. P-50D flange was not examined for safety & ALARA reasons. P-50C was examined from a distance (note: from 625' floor near pump). No evidence of leakage was recorded.

Refueling Outage #14; December 1999

Refueling outage pressure test RT-71A did not record any indication of leakage or evidence of leakage in the area of interest for pump 50C.

September 6, 2000

The System Engineer initiated work order 24013898 to clean boric acid from the CCW pipe flange in the area of interest for pump 50C. Activities were completed April 8, 2001. The CCW flange was determined to be in good condition. No cleaning or inspection was performed on the pump casing.

Refueling Outage #15; May 2001

Refueling outage pressure test RT-71A recorded an indication of evidence of leakage in the area of interest for pump 50C. However, no active steam plume or water dripping was seen.

Forced Outage 01-5001, December 21, 2001

During Health Physics activities in the area of P-50C a boron accumulation was discovered at Component Cooling Water (CCW) flange located in the cooling water piping to P-50C (see CPAL0104122). After cleaning and inspecting, stud wastage was recorded (WO 24114404). No active leakage was seen.

Attach # 11
EA-CPAL 0104122-01

Test Ref.: "F"

Test Type: Immersion – Aerated Boric Acid

Org/Date: Brookhaven National Laboratory (1982)

Reference: "Boric Acid Corrosion of Ferritic Reactor Components." Brookhaven National Laboratory, July 1982, NUREG/CR-2827 (9).

Test Configuration, Procedure, and Results

The testing performed by BNL was not described in any of the references. However, it can be inferred that A-193 Grade B7, AISI 4130 and 4135 low alloy steel specimens were immersed in an aerated environment containing borated water at various temperatures. The average corrosion rates for these materials were then determined for test durations of 350–1300 hours.

Tests were conducted in solutions with 4,000 ppm boron and 4,000 ppm boron plus LiOH to bring the solution to pH 7.3 which is close to typical PWR operating conditions.

The test cases and key test results are as follows:

Temperature (°F)	Average Corrosion Rate (in/yr)	
	4000 ppm boron	4000 ppm boron + LiOH to 7.3 pH
212	0.108 - 0.124	0.112 - 0.130
352	0.042 - 0.050	0.046 - 0.054
600	0.022 - 0.028	no data

Key observations from these tests are:

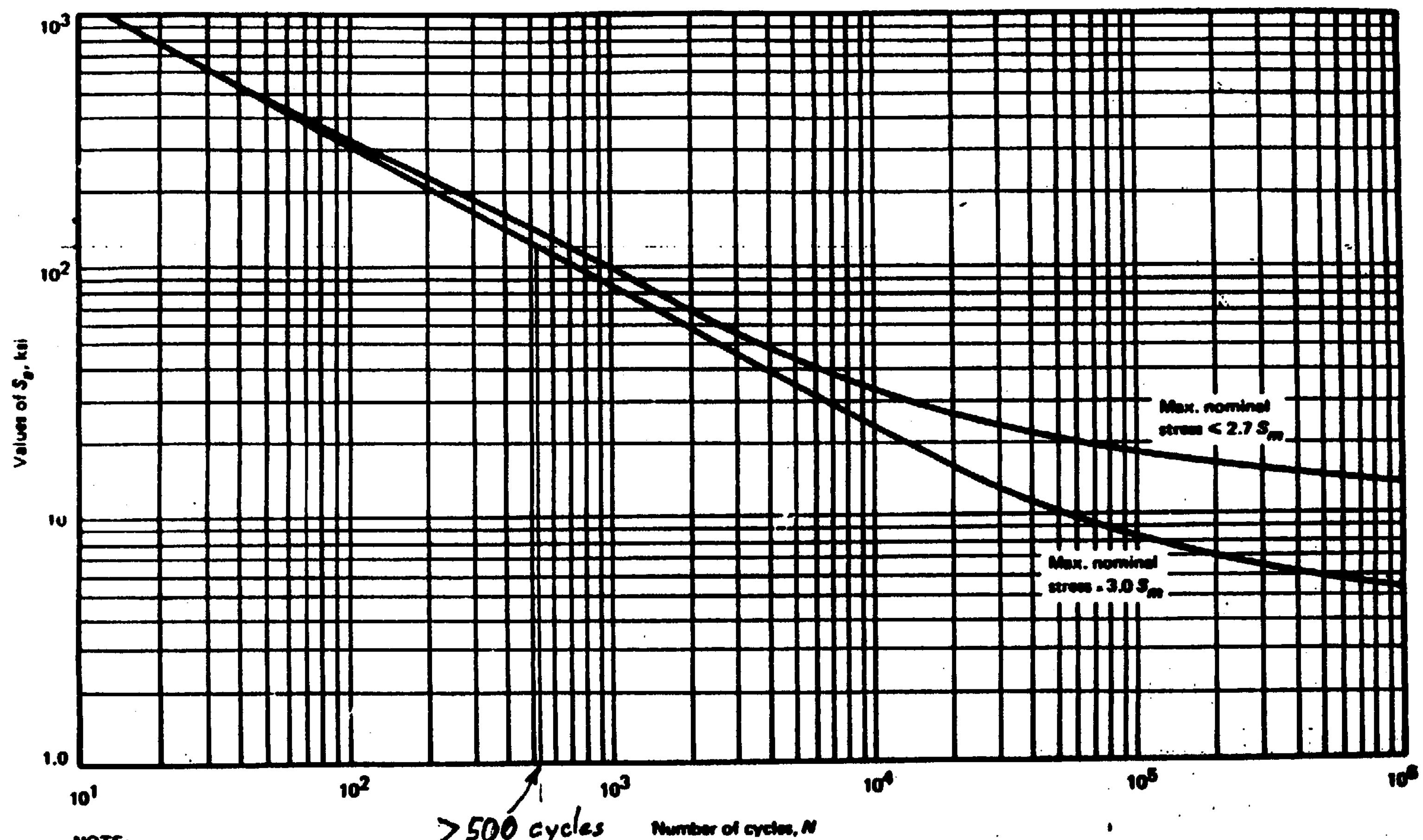
- The corrosion rate decreases for temperatures above 212°F.
- Lithiated primary water has essentially the same high temperature corrosion rate as non lithiated water. This result is not consistent with testing conducted by others at lower temperatures. BNL hypothesizes that the increased ionic species present at higher temperatures increase reaction kinetics over that at lower temperatures.

Conclusions

The main conclusion from these tests is that corrosion rates in aerated water near typical operating boric acid concentrations can be as high as 0.13 in/yr. Work by others has shown that this corrosion rate can increase significantly at higher concentrations.

EPRI 'NMAC' Boric Acid Corrosion Guidebook TR-102745
April 95

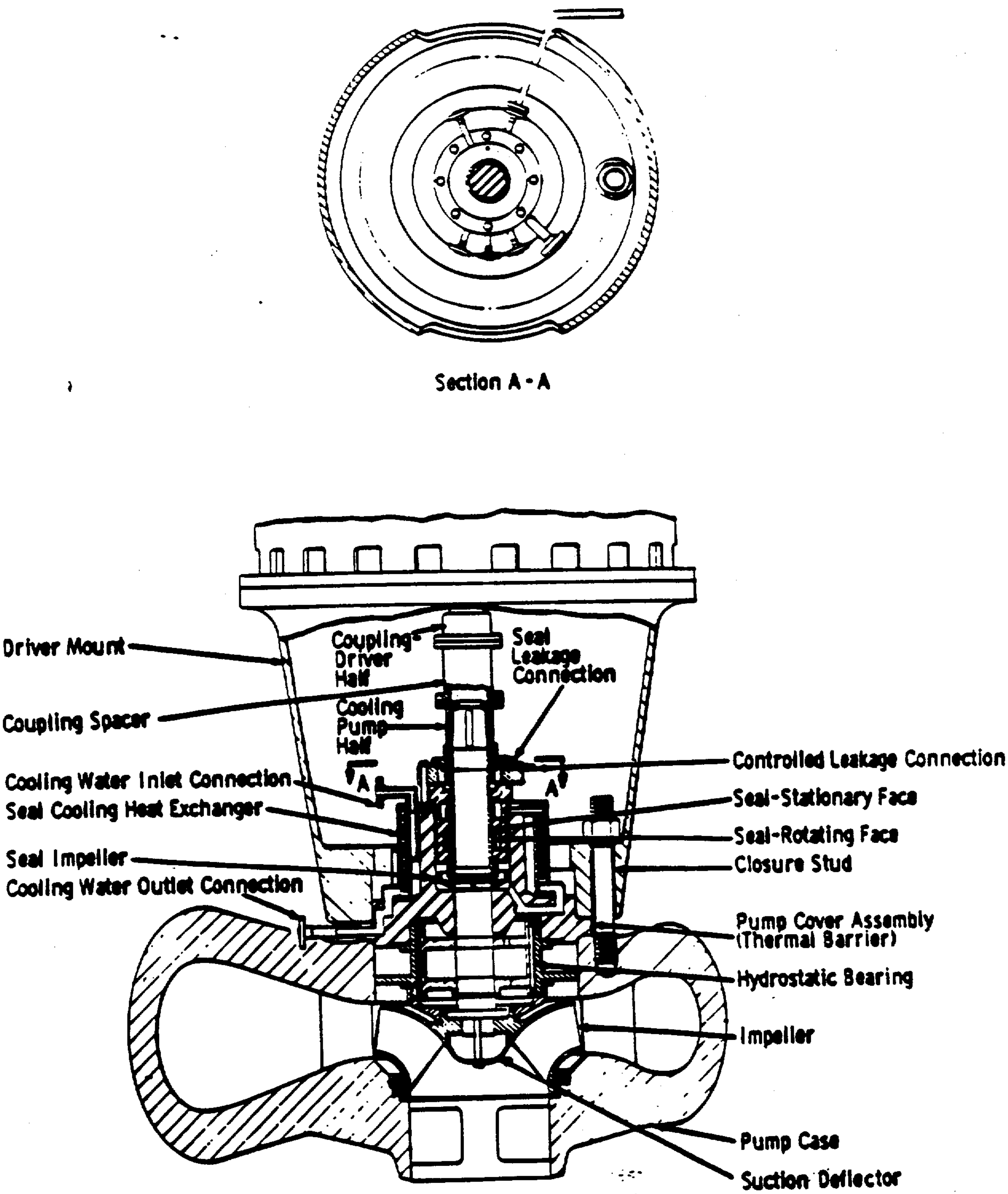
EA-CPAL 0104122-01
Attachment 12
Page 1 of 1



NOTE:
E = 30 X 10^6 psi

FIG. I-9.4 DESIGN FATIGUE CURVES FOR HIGH STRENGTH STEEL BOLTING FOR TEMPERATURES NOT EXCEEDING 700°F
Table I-9.1 Contains Tabulated Values and a Formula for Accurate Interpolation of These Curves

Attach 13 sh 1 of 2
EA-CPAL 0104122-01



CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

PRIMARY COOLANT CIRCULATING PUMP

FIGURE NO 4-4

REVISION NO 0

Attach 13 sh 2 of 2
EA-CPAL 0104122-01