Entergy

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PNP 2015-043

July 7, 2015

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

SUBJECT: Relief Request Number RR 4-22 – Proposed Alternative, Use of Alternate ASME Code Case N-770-1 Baseline Examination

Palisades Nuclear Plant
Docket 50-255
Renewed Facility Operating License No. DPR-20

REFERENCES:

1. Entergy Nuclear Operations, Inc. letter PNP 2014-015, "Relief Request Number RR 4-18 – Proposed Alternative, Use of Alternate ASME Code Case N-770-1 Baseline Examination," dated February 25, 2014 (ADAMS Accession No. ML14056A533).

2. NRC electronic mail, "Palisades Nuclear Plant – Verbal Authorization for Relief Request RR 4-18 - MF3508," dated March 13, 2014 (ADAMS Accession No. ML14073A274).

3. NRC letter, "Palisades Nuclear Plant – Proposed Alternative, Use of Alternate ASME Code Case N-770-1 Baseline Examination (TAC No. MF3508)," dated September 4, 2014 (ADAMS Accession No. ML14223B226).

4. Entergy Nuclear Operations, Inc. letter PNP 2015-037, "Relief Request Number RR 4-21 – Proposed Alternative, Use of Alternate ASME Code Case N-770-1 Baseline Examination," dated May 22, 2015 (ADAMS Accession No. ML15147A616).

Dear Sir or Madam:

Pursuant to 10 CFR 50.55a(z)(2), Entergy Nuclear Operations, Inc. (ENO) hereby requests NRC approval of the Request for Relief for a Proposed Alternative for the Palisades Nuclear Plant (PNP).

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NRR

In Reference 1, ENO requested Nuclear Regulatory Commission (NRC) approval of a proposed alternative described in relief request number RR 4-18 for PNP. The request was associated with the use of an alternative to the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Code Case N-770-1, as conditioned by 10 CFR 50.55a(g)(6)(ii)(F)(1) and 10 CFR 50.55a(g)(6)(ii)(F)(3), dated June 21, 2011. In Reference 2, the NRC staff verbally authorized on March 12, 2014 the use of relief request RR 4-18 at PNP until the next refueling outage, scheduled to start in fall 2015. In Reference 3, a NRC safety evaluation detailing the technical basis for the verbal authorization was issued on September 4, 2014. Subsequent to Reference 3, a discrepancy was discovered in one of the calculations that supported relief request RR 4-18. In Reference 4, ENO requested NRC approval of a proposed alternative described in relief request number RR 4-21, which would supersede relief request number RR 4-18 upon approval.

This submittal requests approval of the proposed alternative described in the enclosed relief request number RR 4-22. This relief request is associated with the use of an alternative to the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Code Case N-770-1, as conditioned by 10 CFR 50.55a(g)(6)(ii)(F)(1) and 10 CFR 50.55a(g)(6)(ii)(F)(3).

The proposed duration of this relief request is to the end of the current fourth 10-year interval and through the first refueling outage in the fifth 10-year interval, scheduled for spring 2017. The fourth interval will end on December 12, 2015 and the fifth interval will begin on December 13, 2015.

ENO requests NRC approval by September 20, 2015, to support planning for the fall 2015 refueling outage and the spring 2017 refueling outage.

This submittal contains no proprietary information.

This letter contains no new commitments and no revised commitments.

Sincerely,

jah/jse

Enclosure: Entergy Nuclear Operations, Inc., Palisades Nuclear Plant, Relief Request Number RR 4-22 Proposed Alternative

cc: Administrator, Region III, USNRC
Project Manager, Palisades, USNRC
Resident Inspector, Palisades, USNRC

ENCLOSURE

ENTERGY NUCLEAR OPERATIONS, INC.
PALISADES NUCLEAR PLANT

RELIEF REQUEST NUMBER RR 4-22 PROPOSED ALTERNATIVE
in Accordance with 10 CFR 50.55a(z)(2)
Hardship Without a Compensating
Increase in Level of Quality and Safety

1. ASME CODE COMPONENT(S) AFFECTED / APPLICABLE CODE EDITION

Components / Numbers:	See Enclosure Table 1 Pressure Retaining Dissimilar Metal Piping Butt Welds Containing Alloy 82/182
Code of Record:	For the fourth interval, the American Society of Mechanical Engineers (ASME) Section XI, 2001 Edition through 2003 Addenda as amended by 10 CFR 50.55a For the fifth interval, ASME Section XI, 2007 Edition with the 2008 Addenda ASME Code Case N-770-1, "Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated with UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities, Section XI, Division 1"
N-770-1 Inspection Item:	A-2 and B
Description:	Class 1 Pressurized Water Reactor (PWR) pressure retaining Dissimilar Metal Piping and Vessel Nozzle Butt Welds containing Alloy 82/182
Unit / Inspection Interval:	Palisades Nuclear Plant (PNP) / Fourth 10-Year Interval December 13, 2006 through December 12, 2015 and Fifth Interval December 13, 2015 through December 12, 2025

2. APPLICABLE CODE REQUIREMENTS

For the fourth interval, the applicable code is the ASME Boiler and Pressure Vessel Code, Rules for Inservice Inspection of Nuclear Power Plant Components, Section XI, 2001 Edition through 2003 Addenda, as amended by 10 CFR 50.55a. For the fifth interval, the applicable code is the ASME Boiler and Pressure Vessel Code, Rules for Inservice Inspection of Nuclear Power Plant Components, Section XI, 2007 Edition with the 2008 Addenda, as amended by 10 CFR 50.55a.

With the issuance of a revised 10 CFR 50.55a in June 2011, the Nuclear Regulatory Commission (NRC) staff incorporated, by reference, Code Case N-770-1. Specific

implementing requirements are documented in 10 CFR 50.55a(g)(6)(ii)(F) and are listed below:

A. Regulation 10 CFR 50.55a(g)(6)(ii)(F)(1) states "Licensees of existing, operating pressurized water reactors as of July 21, 2011 must implement the requirements of ASME Code Case N-770-1, subject to the conditions specified in paragraphs (g)(6)(ii)(F)(2) through (g)(6)(ii)(F)(10) of this section, by the first refueling outage after August 22, 2011."

B. Regulation 10 CFR 50.55a(g)(6)(ii)(F)(3) states that baseline examinations for welds in Code Case N-770-1, Table 1, Inspection Items A-1, A-2, and B, must be completed by the end of the next refueling outage after January 20, 2012.

The welds covered by this proposed alternative would be classified as Inspection Items A-2 and B (described below) for which visual and essentially 100 percent volumetric examination, as amended by 10 CFR 50.55a(g)(6)(ii)(F)(4), in part, are required, per NRC interpretation.

ASME Code Case N-770-1, Table 1, Examination Categories, as amended by 10 CFR 50.55a(g)(6)(ii)(F)		
CLASS 1 PWR Pressure Retaining Dissimilar Metal Piping and Vessel Nozzle Butt Welds Containing Alloy 82/182		
Parts Examined	Insp Item	Extent and Frequency of Examination
Unmitigated butt weld at Hot Leg operating temperature (-2410) ≤ 625°F (329°C)	A-2	Bare metal visual examination each refueling outage. Essentially 100% volumetric examination for axial and circumferential flaws in accordance with the applicable requirements of ASME Section XI, Appendix VIII, every five years. Baseline examinations shall be completed by the end of the next refueling outage after January 20, 2012.
Unmitigated butt weld at Cold Leg operating temperature (-2410) ≥ 525°F (274°C) and < 580°F (304°C)	B	Bare metal visual examination once per interval. Essentially 100% volumetric examination for axial and circumferential flaws in accordance with the applicable requirements of ASME Section XI, Appendix VIII, every second inspection period not to exceed 7 years. Baseline examinations shall be completed by the end of the next refueling outage after January 20, 2012.

ASME Section XI, Appendix VIII, Supplement 10, "Qualification Requirements for Dissimilar Metal Piping Welds," is applicable to dissimilar metal (DM) welds without cast materials.

3. <u>REASON FOR REQUEST</u>

Examinations of the DM welds listed in Enclosure Table 1 of this request as required by ASME

Code Case N-770-1, and as conditioned by 10 CFR 50.55a(g)(6)(ii)(F), would involve a hardship without a compensating increase in level of quality and safety.

These DM welds are nominal pipe size (NPS) 2 inches and greater, full penetration branch connection welds, installed in primary coolant loop piping. See Attachment 1 for typical configuration.

The relevant conditions for this proposed alternative are ASME Section XI Code Case N-770-1, and 10 CFR 50.55a(g)(6)(ii)(F) items (1) and (3), which address performing the required baseline examinations.

Regulation 10 CFR 50.55a(g)(6)(ii)(F)(1) requires that licensees implement the requirements of ASME Code Case N-770-1, subject to the conditions specified in paragraphs (g)(6)(ii)(F)(2) through (g)(6)(ii)(F)(10) of this section, by the first refueling outage after August 22, 2011.

Regulation 10 CFR 50.55a(g)(6)(ii)(F)(3) requires that baseline examinations for welds in Code Case N-770-1 Table 1, Inspection Items A-1, A-2, and B be completed by the end of the next refueling outage after January 20, 2012.

Relief is requested from 10 CFR 50.55a(g)(6)(ii)(F) items (1) and (3) for performance of required baseline volumetric examinations of the eight cold leg welds and one hot leg weld listed in the Enclosure Table 1.

Hardship

ENO proposes to defer required volumetric examinations of the subject PCS branch connection welds until refueling outage 1R25, scheduled to start in the spring of 2017, on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Due to the location of the subject nine PCS branch connection welds, performing ultrasonic examinations of the welds involve significant radiation exposure to personnel. Total dose incurred by examination, radiation protection, and supervisory personnel during ultrasonic examinations of the nine weld locations is estimated to be 14.5 Rem (see Attachment 1, Table 2). This dose includes preparation activities, and credits dose reduction controls and measures such as shielding, decontamination of components, high efficiency particulate air filter ventilation units, cameras, and remote telemetry. It also includes erecting and removing scaffolding, conducting surveys, removing insulation, etc. The total dose that would be incurred for repair/replacement activities, involving installation of a weld pad with a half nozzle repair at each of the nine weld locations, is estimated to be at least 40 Rem. The dose incurred for each of the locations would range from approximately 2 Rem to 9 Rem.

ENO has implemented a program to reduce PCS collective radiation exposure by increasing PCS coolant pH. Increasing the pH of the PCS will reduce corrosion rates within the PCS, thereby reducing the amount of corrosion products released into the coolant and reducing plant radiation levels. By refueling outage 1R25, this program is expected to have reduced PCS collective radiation exposure by approximately 5%. This would result in about 725 mRem dose savings for ultrasonic examinations of the nine welds, and about 2000 mRem dose savings for repairs of the nine weld locations.

Areva is currently performing a flaw analysis to determine inspection requirements for branch connection welds, which would be adopted in either Code Case N-770, for volumetric examinations, or in Code Case N-722, "Additional Examinations for PWR Pressure Retaining Welds in Class 1 Components Fabricated with Alloy 600/82/182 Materials, Section XI, Division 1," for visual examinations, or be adopted in both N-770 and N-722. The new Code requirements are expected to either reduce the scope of the required volumetric inspections or require only bare metal visual inspections, and would result in considerable radiological dose savings to personnel. The results of the Structural Integrity Associates (SIA) report and calculations in Attachment 2 would support these changes.

Volumetric techniques for this PNP weld joint configuration (see Attachment 1, Figures 1 and 2) are currently under development through the Performance Demonstration Initiative (PDI) qualification program. Under the PDI program, mock-ups have been fabricated in accordance with the PDI specimen fabrication program in order to develop examination techniques. Qualification of procedures and personnel are underway to perform the qualified examinations of these welds. Qualification of the volumetric inspection technique is currently on schedule. However, if there are future delays in the qualification process, then the inspection technique may not be ready for use in the 1R24 refueling outage.

There are no existing ASME Section XI Code rules for a branch connection weld repair, involving installation of a weld pad with a half nozzle repair, but a repair Code Case for a branch connection weld repair is being developed by the ASME Code committee. The Code Case will not likely be issued by the ASME Code Committee by the 1R24 refueling outage, and a repair strategy needs to be approved prior to inspecting the branch connection welds.

Babcock & Wilcox (B&W) full penetration branch welds exist in the industry that have not been placed in service. For information gathering purposes, ENO would prefer to examine these branch welds using the newly qualified UT technique rather than perform a first-time examination technique on an operating unit. However, there is not sufficient time to perform this trial run prior to the PNP refueling outage. At best, the NDE procedures will be ready just prior to the PNP outage. Also, since these are B&W branch connection welds instead of CE branch connection welds, the PNP tooling and procedures may not be suitable for these B&W welds. The B&W NDE qualification will not be complete until next year to examine these branch welds. The branch welds were fabricated by B&W instead of CE but used a similar fabrication process that is more representative of actual Nuclear Steam Supply System fabrication processes with a similar design.

ENO requests relief from the requirements of 10 CFR 50.55a(g)(6)(ii)(F) until refueling outage 1R25. Performing the inspections and implementing any needed repairs during 1R25 would result in personnel radiological dose savings. By 1R25, industry activities will have been completed that are expected to reduce, if not eliminate, the hardship of performing volumetric inspections of PCS branch connection welds. In addition, ENO would prefer to examine these branch welds using the aforementioned Code Cases.

4. PROPOSED ALTERNATIVE AND BASIS FOR USE

Proposed Alternative

1) Perform periodic system leakage tests in accordance with ASME Section XI Examination Category B-P, Table IWB-2500-1.

2) Perform visual examinations (per Code Case N-722-1) and dye penetrant surface examinations (per ASME Section XI Examination Category B-J, Table IWB-2500-1) of the welds in accordance with ASME requirements.

3) Perform a volumetric examination, using ASME Code, Section XI, Appendix VIII, Supplement 10 qualified procedures, equipment and personnel, on each of the nine subject welds of this alternative during planned spring 2017 refueling outage 1R25.

4) Until the next scheduled refueling outage, if unidentified PCS leakage increases by 0.15 gpm above the WCAP-16465NP baseline mean, and is sustained for 72 hours, ENO will take action to be in Mode 3 within 6 hours and Mode 5 within 36 hours, and perform bare metal visual examinations of the nine subject welds of this alternative, unless it can be confirmed that the leakage is not from these welds.

Regulation 10 CFR 50.55a(g)(6)(ii)(F)(1) states "Licensees of existing, operating pressurized water reactors as of July 21, 2011 must implement the requirements of ASME Code Case N-770-1, subject to the conditions specified in paragraphs (g)(6)(ii)(F)(2) through (g)(6)(ii)(F)(10) of this section, by the first refueling outage after August 22, 2011."

Regulation 10 CFR 50.55a(g)(6)(ii)(F)(3) states that baseline examinations for welds in Code Case N-770-1, Table 1, Inspection Items A-1, A-2, and B, must be completed by the end of the next refueling outage after January 20, 2012.

Pursuant to 10 CFR 50.55a(z)(2), ENO will comply with 10 CFR 50.55a(g)(6)(ii)(F) by completing the baseline volumetric examinations as required by Code Case N-770-1, for each of the nine subject welds of this alternative, prior to the end of the refueling outage 1R25, currently scheduled to start in spring 2017.

Basis for Use

Table 1 in Attachment 1 of this enclosure describes the eight cold leg welds and one hot leg weld that have not been examined in accordance with Code Case N-770-1 examination requirements, as required by 10 CFR 50.55a(g)(6)(ii)(F) items (1) and (3).

Examination History

Table 1 in Attachment 1 also provides examination history information for the nine weld locations for which relief is requested. No evidence of through-wall cracking for these components has been identified during these inspections. Moreover, for the three weld locations that were not subject to surface or visual examinations during the 1R23 refueling outage (i.e., weld no.'s 3, 6, and 7 in Table 1), maintenance activities in the vicinity of the weld locations during the 1R23 refueling outage did not identify observations of leakage from the welds.

Structural Evaluation

Background

ENO submitted a proposed alternative (relief request number RR 4-18) concerning volumetric examinations of the subject branch connection welds on February 25, 2014 (Reference 1), and supplemental information was submitted on March 1, March 4, March 6, March 9, and March 11, 2014 (References 2 through 7).

On March 12, 2014 (Reference 8), the NRC staff verbally authorized the use of relief request number RR 4-18 at PNP until the next scheduled refueling outage, scheduled in the fall of 2015 (1R24). A NRC safety evaluation detailing the technical basis for the verbal authorization was subsequently issued on September 4, 2014 (Reference 9).

Subsequent to Reference 9, a discrepancy was discovered in one of the calculations that supported relief request RR 4-18. In Reference 10, ENO requested NRC approval of a proposed alternative described in relief request number RR 4-21, which would supersede relief request number RR 4-18 upon approval.

Structural Evaluation Using Finite Element Analysis Methodology

The technical basis for the proposed alternative is provided in Attachment 2 of this enclosure, which contains the following report and calculations:

SI Report No. 1400669.401.R0, "Evaluation of the Palisades Nuclear Plant Branch Line Nozzles for Primary Water Stress Corrosion Cracking," Revision 0, dated May 14, 2015.

SI Calculation, File No. 1400669.313, "Crack Growth Analysis of the Hot Leg Drain Nozzle," Revision 0, dated May 11, 2015.

SI Calculation, File No. 1400669.323, "Crack Growth Analysis of the Cold Leg Bounding Nozzle," Revision 0, dated May 11, 2015.

SI Calculation, File No. 1400669.310, "Finite Element Model for Hot Leg Drain Nozzle," Revision 0, dated March 9, 2015.

SI Calculation, File No. 1400669.320, "Finite Element Model Development for the Cold Leg Drain, Spray, and Charging Nozzles," Revision 0, dated April 3, 2015.

SI Calculation, File No. 1400669.312, "Hot Leg Drain Nozzle Weld Residual Stress Analysis," Revision 0, dated May 5, 2015.

SI Calculation, File No. 1400669.322, "Cold Leg Bounding Nozzle Weld Residual Stress Analysis," Revision 0, dated May 5, 2015.

In these calculations, SI used a finite element analysis (FEA) approach to evaluate postulated flaws in the hot leg and cold leg nozzles. These models were used to perform weld residual stress evaluations and calculations of stress intensity factors in the DM welds. Utilizing these new stress intensity factor distributions for postulated circumferential and axial flaws in the DM welds, crack growth due to PWSCC was evaluated for both the hot leg and cold leg configurations. Crack growth durations were then plotted on charts to show the

service life of the hot leg and cold leg configurations based on crack growth from an assumed initial flaw depth of 0.025 inch. It should be noted that PWSCC was the only crack growth mechanism considered in this evaluation (i.e., PWSCC growth of a postulated axial and circumferential flaw in the weld).

Using the FEA approach, the calculations determined that, for the hot leg drain nozzle, the time for an initial 0.025 inch deep flaw to grow to 75% through-wall was calculated to be 30.5 years for the bounding axial flaw (36.7 years to go 95% through-wall) and 33.9 years for the circumferential flaw (42.1 years to go 95% through-wall).

For the cold leg drain nozzle, the time for an initial 0.025 inch deep flaw to grow to 75% through-wall is 64.5 years for the bounding axial flaw (77 years to go 95% through-wall) and 55.6 years for the circumferential flaw (66.2 years to go 95% through-wall).

By the 1R25 refueling outage, PNP will have operated for 28.8 effective full power years (Reference 22).

The SI report in Attachment 2 discusses a benefit in crack growth rate when Alloy 182 weld metal underwent post weld heat treatment (PWHT). This benefit ranged from a factor of two to four. Dominion Engineering recommended that crack growth rates be reduced by a conservative factor of two based on this data. Using this factor for the nozzles at PNP will increase the crack growth times previously discussed. For comparison, only the bounding crack growth life will be discussed here. This is the axial flaw for the hot leg drain nozzle, and the circumferential flaw for the cold leg nozzles. A factor of two will increase the limiting hot leg drain nozzle duration to 61.0 years, and the bounding cold leg nozzle to 111.2 years, for flaws to grow 75% through wall. Similarly, the durations for 95% through-wall flaws would increase to 73.4 years for the hot leg drain nozzle and 132.4 years for the cold leg nozzle.

Attachment A of the SI Report No. 1400669.401.R0 in Attachment 2 includes a calculation of the initiation time for the hot leg drain nozzle and cold leg bounding nozzle using one of the models in the xLPR program that was developed by the Electric Power Research Institute (EPRI). The calculation of initiation time based on the PNP results shows that the time to initiation for the hot leg drain nozzle is approximately 130 years. If the crack initiation time is combined with the PWHT reduction in crack growth rate, the time for a flaw to grow to 75% through-wall would approach 200 years for the hot leg drain nozzle and would exceed 200 years for 95% through-wall. Using the crack initiation value for the cold leg nozzles, the time for a flaw to grow to 75% through-wall would exceed 600 years. For both the hot leg and cold leg nozzles, the time for a flaw to grow to 75% through-wall is well beyond the life of the plant.

As documented in Attachment 3, Areva, Inc. performed an independent review of the SI report and calculations. The review concluded that the assumptions in the SI documents are conservative and appropriate, and support this relief request to defer volumetric examinations of the subject welds to the planned spring 2017 refueling outage, as described in the Proposed Alternative above. Further, based on projected durations of approximately 191 years for a hot leg flaw to reach 75% through wall, more than 200 years for a hot leg flaw to reach 95% through wall, and greater than 600 years for a cold leg flaw to reach 75% through wall, the review also concluded that bare metal visual examination is sufficient to ensure that cracking in the hot and cold leg nozzles would not pose an immediate safety concern for PNP.

ENO understands that these results will be shared with ASME to assist in developing inspection requirements and inspection intervals for this category weld in either Code Case N-722 (bare metal) or Code Case N-770 (volumetric), and with these results, it is likely that the ASME Section XI TG - HSNA (Task Group - High Strength Nickel Alloy) will recommend revisions to Code Case N-722.

Based on these calculations and the hardship discussion, complying with the specified examination requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Additional Information

Post Weld Heat Treatment

All welds referenced in this relief request were post weld heat treated at the vendor's facility during original fabrication. Per Combustion Engineering detail weld procedure (MA-41, Revision 0), heat treatment of the hot leg nozzle weld (weld number 5-675) consisted of an intermediate post weld heat treatment at 1100°F, - 0°F/+ 50°F, for 15 minutes. Final heat treatment at the hot leg nozzle consisted of 1150°F, +/- 25°F, for one hour per inch thickness of weld. No heat treatment was performed at the site for the subject welds.

The post weld heat treatment was applied to the nozzle welds by heating the affected nozzle/weld as part of the hot leg pipe assembly in a furnace. The hot leg drain nozzle (piece number 675-03) was installed into the hot leg pipe (pipe assembly 673-04) by cutting a hole into the hot leg piping and then fitting and welding the hot leg drain nozzle in place using an intermediate heat treat of 1100°F + 50°F for 15 minutes. The backing ring for the weld was removed, the weld was dye penetrant tested, and then back welded. The back weld was dye penetrant tested, and any indications were repaired, and then a final dye penetrant test was performed. The weld was then back clad and dye penetrant tested. Any indications were removed and dye penetrant tested again. The hot leg drain nozzle weld was radiographed and any defects were removed, weld repaired, and then dye penetrant tested and radiographed. The pipe assembly with the hot leg drain nozzle installed was post weld heat treated in a furnace using a post weld heat treat of 1150°F ± 25°F, holding this temperature for one hour/inch thickness of weld.

Weld Repair History

The manufacturing/quality plan provided in the specification for the PCS piping provides instructions for performing weld repairs based on the results of NDE testing. Any defects identified in the nozzle welds would have been removed prior to final furnace heat treatment of the assembly. PNP is unable to locate post-fabrication documentation other than the weld radiographs taken after the final furnace heat treatment. These radiographs represent the condition of the subject welds at the time of installation at the site. A search of PNP records did not identify any repairs performed on the subject welds since installation.

Weld and Cladding Geometry

The hot leg weld is ½ inch wide at the ID and then increases in width at a 7-½° (+2°/-0°) angle to the OD of the hot leg pipe. The thickness of the hot leg pipe is 3-¾ inches. See Reference 2.

The thickness of the alloy 182/82 cladding on the inside surface of the weld is 1/4 inch nominal thickness with a minimum thickness of 5/32 inch. The inner radius at the hot leg drain is 21.34 inches measured from the inside surface of the cladding to the centerline of the hot leg pipe.

Detailed information concerning the weld geometry is provided in References 2 and 5.

Basis for Assuming No Weld Repair

The presence of an initial weld repair from plant construction (e.g., extending 50% of the wall thickness from the inside diameter (ID)) is often assumed when modeling Alloy 82/182 piping butt welds. Often for piping butt welds, the residual stress calculated for the ID is a small tensile value, or even compressive, in the absence of an assumed weld repair. In such cases, the possibility of a significant weld repair being present on the weld ID can have a relatively large effect on the calculated stresses, especially on and near the ID surface.

However, for the Alloy 82/182 branch connection welds at PNP, there are two reasons why it is not necessary to include a weld repair assumption in the analysis. First, the design for this weldment specifies a 360° backweld on the ID surfaces of the pipe that is about 0.25 inch thick. This design feature results in elevated residual stress levels at the ID surface prior to the PWHT being applied. The residual stress levels at the inside surface due to the presence of the backweld are similar to what would be expected due to the presence of a weld repair on the ID surface.

Second, any weld repairs would have been made prior to PWHT being applied, and would be expected to extend over a relatively limited circumferential portion of the original weld. Similar to the situation for the elevated residual stresses due to the presence of the backweld, the PWHT would relax the residual stresses in the weld repair area, including the substantial relaxation expected at the surface exposed to primary coolant. Moreover, in the unlikely case that initiation occurred in the area of a weld repair, the weld repair would be an additional source of non-axisymmetric crack loading that would tend to drive crack growth in the through-wall direction over a relatively local circumferential region, ultimately resulting in detection of leakage prior to the possibility of unstable pipe rupture.

Basis for Five-Cycle Shakedown Assumption

Operational cycles are frequently included in welding residual stress calculations as a part of determining the operating stress condition. In particular, the standard modeling practice adopted by the xLPR (Extremely Low Probability of Rupture) welding residual stress team, which includes the NRC, national laboratories, and industry participants, specifies that the welded configuration should be cycled between operating conditions and residual conditions to shake down the nonlinear material hardening behavior. Typically, three to five cycles are used to shake down the material's behavior.

Since the primary interest from the residual stress analysis is to provide residual stresses for calculating stress corrosion crack growth under normal operating conditions, it is desirable to determine a stabilized residual stress state that will not change under normal operating cycles. The as-welded residual stresses usually contain localized peak stresses at some nodal locations. Applying a few operating cycles will stabilize the stress peaks and valleys due to the slight stress redistribution at elevated temperatures. It has been determined

through experience that the residual stresses will stabilize after three to five cycles. Five cycles were used for conservatism.

See Reference 2 for additional information.

Operating Conditions

The operating temperature of a component is a primary factor influencing the initiation of PWSCC. Research by EPRI (Reference 18) indicates that the difference in the operating temperature between hot leg locations and cold leg locations is sufficient to significantly influence the time to initiation of PWSCC, with the susceptibility increasing with temperature. The research reports PWSCC is less likely to occur in cold leg temperature penetrations.

All but one of the welds covered by this relief are found in lower temperature regions of the system, typically at temperatures near to T_{cold}, which is approximately 537°F. This means, for these welds, there is a lower probability of crack initiation, and a slower crack growth rate (Reference 19).

Leakage Detection Capabilities

The leak detection methodology presently used by industry is very sensitive. After a number of recent operating events, the industry imposed an NEI 03-08, "Guideline for the Management of Materials Issues," requirement to improve leak detection capability. As a result, virtually all pressurized water reactors (PWRs) in the United States, including PNP, have a leak detection capability of less than or equal to 0.1 gpm (Reference 17). All plants, including PNP, also monitor seven-day moving averages of reactor coolant system leak rates.

Action response times following a detected primary coolant system leak vary, based on the action level exceeded and whether containment entry is required to identify the source of the leak.

Action levels have been standardized for all PWRs, and are based on deviations from:

- the seven day rolling average,
- specific values, and
- the baseline mean.

Leak rate action levels are identified in Pressurized Water Reactor Owners Group (PWROG) report, WCAP-16465, and are stated below:

> Each PWR utility is required to implement the following standard action levels for reactor coolant system (RCS) inventory balance in their RCS leakage monitoring program.
>
> Action levels on the absolute value of unidentified RCS inventory balance (from surveillance data):
>
> Level 1 - One seven day rolling average of unidentified RCS inventory balance values greater than 0.1 gpm.

Level 2 - Two consecutive unidentified RCS inventory balance values greater than 0.15 gpm.

Level 3 - One unidentified RCS inventory balance value greater than 0.3 gpm.

Note: Calculation of the absolute RCS inventory balance values must include the rules for the treatment of negative values and missing observations.

Action levels on the deviation from the baseline mean:

Level 1 - Nine consecutive unidentified RCS inventory balance values greater than the baseline mean [μ] value.

Level 2 - Two of three consecutive unidentified RCS inventory balance values greater than [$\mu + 2\sigma$], where σ is the baseline standard deviation.

Level 3 - One unidentified RCS inventory balance value greater than [$\mu + 3\sigma$].

These action levels have been incorporated into PNP procedures.

A small steam leak from a weld flaw would, over time, result in a rise in containment sump level rate of increase. Containment sump level is continually monitored, and if a rise in the rate of containment sump level increase is observed, plant procedures direct plant operators to identify the source of the leakage. Operators may also be alerted to a leak from a flaw by containment radiation monitoring instrumentation. This instrumentation, required by the Technical Specifications, is capable of detecting a 100 cm^3/min leak in 45 minutes, based on 1% failed fuel. Periodic system leakage tests are performed in accordance with ASME Section XI. Operator walkdowns of containment are periodically performed during power operations at lower levels of containment to detect leakage.

Therefore, with the periodic system leakage tests, the visual and surface examinations performed during 2012 and 2014 refueling outages, the results of the SI evaluation, containment monitoring activities, and the testing, examination, and PCS leakage requirements in the proposed alternative, an acceptable level of quality and safety is provided for identifying degradation from PWSCC prior to a safety-significant flaw developing.

5. DURATION OF PROPOSED ALTERNATIVE

The duration of the proposed alternative is until refueling outage 1R25, which is currently scheduled in spring 2017.

6. REFERENCES

1. Entergy Nuclear Operations, Inc. letter PNP 2014-015, "Relief Request Number RR 4-18 Proposed Alternative, Use of Alternate ASME Code Case N-770-1 Baseline Examination," dated February 25, 2014 (ADAMS Accession No. ML14056A533).

2. Entergy Nuclear Operations, Inc. letter PNP 2014-021, "Response to Request for Additional Information dated February 26, 2014, for Relief Request Number RR 4-18 –

Proposed Alternative, Use of Alternate ASME Code Case N-770-1 Baseline Examination," dated March 1, 2014 (ADAMS Accession No. ML14072A361).

3. Entergy Nuclear Operations, Inc. letter PNP 2014-022, "Response to Second Request for Additional Information dated February 27, 2014, for Relief Request Number RR 4-18 – Proposed Alternative, Use of Alternate ASME Code Case N-770-1 Baseline Examination," dated March 4, 2014 (ADAMS Accession No. ML14063A089).

4. Entergy Nuclear Operations, Inc. letter PNP 2014-028, "Supplemental Response to Request for Additional Information dated February 26, 2014, for Relief Request Number RR 4-18 – Proposed Alternative, Use of Alternate ASME Code Case N-770-1 Baseline Examination," dated March 6, 2014 (ADAMS Accession No. ML14066A409).

5. Entergy Nuclear Operations, Inc. letter PNP 2014-029, "Response to Third Request for Additional Information dated March 5, 2014, for Relief Request Number RR 4-18 – Proposed Alternative, Use of Alternate ASME Code Case N-770-1 Baseline Examination," dated March 6, 2014 (ADAMS Accession No. ML14070A182).

6. Entergy Nuclear Operations, Inc. letter PNP 2014-030, "Second Supplemental Response to Request for Additional Information dated February 26, 2014, for Relief Request Number RR 4-18 – Proposed Alternative, Use of Alternate ASME Code Case N-770-1 Baseline Examination," dated March 9, 2014 (ADAMS Accession No. ML14069A004).

7. Entergy Nuclear Operations, Inc. letter PNP 2014-031, "Response to Fourth Request for Additional Information dated March 11, 2014, for Relief Request Number RR 4-18 – Proposed Alternative, Use of Alternate ASME Code Case N-770-1 Baseline Examination," dated March 11, 2014 (ADAMS Accession No. ML14070A477).

8. NRC Electronic Mail, "Palisades Nuclear Plant – Verbal Authorization for Relief Request RR 4-18 - MF3508," March 13, 2014 (ADAMS Accession No. ML14073A274).

9. NRC letter, "Palisades Nuclear Plant – Proposed Alternative, Use of Alternate ASME Code Case N-770-1 Baseline Examination (TAC No. MF3508)," dated September 4, 2014 (ADAMS Accession No. ML14223B226).

10. Entergy Nuclear Operations, Inc. letter PNP 2015-037, "Relief Request Number RR 4-21 – Proposed Alternative, Use of Alternate ASME Code Case N-770-1 Baseline Examination," May 22, 2015 (ADAMS Accession No. ML15147A616)

11. 10 CFR 50.55a, "Codes and standards," December 11, 2014.

12. ASME Section XI, "Rules For Inservice Inspection of Nuclear Power Plant Components," 2001 Edition with Addenda through 2003.

13. ASME Section XI, Division 1, Code Case N-460, "Alternative Examination Coverage for Class 1 and Class 2 Welds, Section XI, Division 1."

14. Material Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guideline (MRP-139), Revision 1, EPRI, Palo Alto, CA, 2008 (ADAMS Accession No. ML1009700671).

15. Nondestructive Evaluation: Procedure for Manual Phased Array Ultrasonic Examination of Dissimilar Metal Welds, EPRI-DMW-PA-1, Revision 3, 1016645, EPRI Palo Alto, CA, 2008.

16. "Changing the Frequency of Inspections for PWSCC Susceptible Welds at Cold Leg Temperatures", in Proceedings of 2011 ASME Pressure Vessels and Piping Conference, July 17-21, 2011, Baltimore, MD.

17. WCAP-16465-NP, Rev. 0, "Pressurized Water Reactor Owners Group Standard RCS Leakage Action Levels and Response Guidelines for Pressurized Water Reactors," Westinghouse Electric Co., September 2006 (ADAMS Accession No. ML070310082).

18. Electric Power Research Institute: PWSCC of Alloy 600 Materials in PWR Primary System Penetrations, EPRI, Palo Alto, CA, 1994, TR-103696 (ADAMS Accession No. ML013110446).

19. Materials Reliability Program Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 82, 182, and 132 Welds (MRP-115), EPRI, Palo Alto, CA, 2004, 1006696 (ADAMS Accession No. ML051100204).

20. PNP Technical Specification Surveillance Procedure RT-71A, "Primary Coolant System, Class 1 System Leakage Test," Revision 18.

21. Pressurized Water Reactor (PWR) Owner's Group Letter OG-12-89, "Transmittal of 'Final Relief Request Framework' under Relief Request for Large Diameter Cold Leg Locations with Obstructions (PA-MSC-0934)," March 8, 2012.

22. WCAP-15353-Supplement 1-NP, Palisades Reactor Pressure Vessel Fluence Evaluation, Revision 0, May 2010 (ADAMS Accession Number ML110060692).

7. <u>ATTACHMENTS</u>

Attachment 1

Table 1 - Weld Examination History

Table 2 – Dose Estimate for Weld Prep and Examinations of Alloy 600 Welds with Additional Dose Reduction Planning Implemented

Figure 1 - Nozzle Assembly Materials

Figure 2 - Hot Leg Drain Nozzle Configuration (Representative)

Attachment 2

1. Structural Integrity Associates, Inc. Report No. 1400669.401.R0, "Evaluation of the Palisades Nuclear Plant Branch Line Nozzles for Primary Water Stress Corrosion Cracking," Revision 0, dated May 14, 2015.

2. Structural Integrity Associates, Inc. Calculation, File No. 1400669.313, "Crack Growth Analysis of the Hot Leg Drain Nozzle," Revision 0, dated May 11, 2015.

3. Structural Integrity Associates, Inc. Calculation, File No. 1400669.323, "Crack Growth Analysis of the Cold Leg Bounding Nozzle," Revision 0, dated May 11, 2015.

4. Structural Integrity Associates, Inc. Calculation, File No. 1400669.310, "Finite Element Model for Hot Leg Drain Nozzle," Revision 0, dated March 9, 2015.

5. Structural Integrity Associates, Inc. Calculation, File No. 1400669.320, "Finite Element Model Development for the Cold Leg Drain, Spray, and Charging Nozzles," Revision 0, dated April 3, 2015.

6. Structural Integrity Associates, Inc. Calculation, File No. 1400669.312, "Hot Leg Drain Nozzle Weld Residual Stress Analysis," Revision 0, dated May 5, 2015.

7. Structural Integrity Associates, Inc. Calculation, File No. 1400669.322, "Cold Leg Bounding Nozzle Weld Residual Stress Analysis," Revision 0, dated May 5, 2015.

Attachment 3

Areva, Inc. letter number AREVA-15-02279, "Transmittal of Updated Task 6 Letter to Support Palisades Relief Request for PA-MSC-1283," dated July 7, 2015.

ATTACHMENT 1

Table 1
Weld Examination History

No.	Description	ISI Weld ID	Location	1R19 Examinations	1R20 Examinations	1R21 Examinations	1R22 Examinations	1R23 Examinations
1.	2 inch Cold Leg Charging Nozzle	PCS-30-RCL-1A-11/2	P-50A Discharge Leg	Visual (Report# 4046 Exam number 06-26)				Surface (Report# 1R23-PT-14-025)
2.	2 inch Cold Leg Drain Nozzle	PCS-30-RCL-1A-5/2	P-50A Suction Leg	Visual (Report# 4047 Exam number 07-28.1)				Surface (Report# 1R23-PT-14-031)
3.	3 inch Cold Leg Pressurizer Spray Nozzle	PCS-30-RCL-1B-10/3	P-50B Discharge Leg			Visual (Report# VT-10-069)	Surface (Report# 1R22-PT-12-039)	
4.	2 inch Cold Leg Drain Nozzle	PCS-30-RCL-1B-5/2	P-50B Suction Leg			Visual (Report# VT-10-048)		Surface (Report #1R23-PT-14-032)
5.	2 inch Cold Leg Charging Nozzle	PCS-30-RCL-2A-11/2	P-50C Discharge Leg		Visual (Report# VT-09-083)			Surface (Report# 1R23-PT-14-019)
6.	3 inch Cold Leg Pressurizer Spray Nozzle	PCS-30-RCL-2A-11/3	P-50C Discharge Leg		Visual (Report# VT-09-035)		Surface (Report# 1R22-PT-12-032)	
7.	2 inch Cold Leg Drain Nozzle	PCS-30-RCL-2A-5/2	P-50C Suction Leg		Visual (Report# VT-09-038)			
8.	2 inch Cold Leg Drain and Letdown Nozzle	PCS-30-RCL-2B-5/2	P-50D Suction Leg			Visual (Report# VT-10-071)		Surface (Report# 1R23-PT-14-020)
9.	2 inch Hot Leg Drain Nozzle	PCS-42-RCL-1H-3/2	A Hot Leg	Visual (Report# 4047 Exam number 07-23.1)	Visual (Report# VT-09-062)	Visual (Report# VT-10-022)	Visual (Report# 1R22-VT-12-076)	Visual (Report# 1R23-VT-14-059)

ATTACHMENT 1

Table 2

Dose Estimate for Weld Prep and Examinations of Alloy 600 Welds with Additional Dose Reduction Planning Implemented

	DESCRIPTION	Area	ISI WELD ID	DOSE RATES (mrem/hr)				Profile		Detection		TWS		Scaffold		Insulation		TOTAL DOSE
				CONTACT	12 inch	G/A	LDWA	HOURS	DOSE	HOURS	DOSE	HOURS	DOSE	HOURS	DOSE	HOURS	DOSE	
1	2" Cold Leg Charging Nozzle	P-50A	PCS-30-RCL-1A-11/2	120	70	30	5	1	155	0.75	105	1	140	11	495	6	180	1075
2	2" Cold Leg Drain Nozzle	P-50A	PCS-30-RCL-1A-5/2	400	50	20	5	1	425	0.75	75	1	100	0	0	6	120	720
3	3" Cold Leg PZR Spray Nozzle	P-50B	PCS-30-RCL-1B-10/3	900	150	80	20	1	1000	0.75	225	1	300	6	700	6	450	2675
4	2" Cold Leg Drain Nozzle	P-50B	PCS-30-RCL-1B-5/2	500	60	40	20	1	560	0.75	90	1	120	0	0	6	250	1020
5	2" Cold Leg Charging Nozzle	P-50C	PCS-30-RCL-2A-11/2	80	50	20	20	1	120	0.75	75	1	100	6	420	6	150	865
6	3" Cold Leg PZR Spray Nozzle	P-50C	PCS-30-RCL-2A-11/3	900	155	80	23	1	1003	0.75	233	1	310	6	700	6	250	2496
7	2" Cold Leg Drain Nozzle	P-50C	PCS-30-RCL-2A-5/2	400	65	30	23	1	453	0.75	98	1	130	0	0	6	250	931
8	2" Cold Leg Drain / Letdown Nozzle	P-50D	PCS-30-RCL-2B-5/2	500	100	100	35	1	635	0.75	150	1	200	0	0	6	450	1435
9	2" Hot Leg Drain Nozzle	A S/G	PCS-42-RCL-1H-3/2	140	80	60	50	1	250	0.75	120	1	160	0	0	7	350	880
									4601		1170		1560		2315		2450	**12096**

Profile - 1 person at the weld, 1 person in the G/A, 1 firewatch in the LDWA

Detection - Requires 2 people at the weld or within arms length of the weld

TWS - Requires 2 people at the weld or within arms length of the weld

Superviser Oversite - 1 person at all times in G/A to LDWA 1500

RP Technician (8% of prep and inspection dose) 968

Scaffold and Insulation dose taken from work performed in 1R23 as tracked by work order

Dose Rate data extracted from surveys PLP-1401-0291, PLP-1401-0459, PLP-1401-0275, PLP-14-2-0500 and from dose reduction planning data.

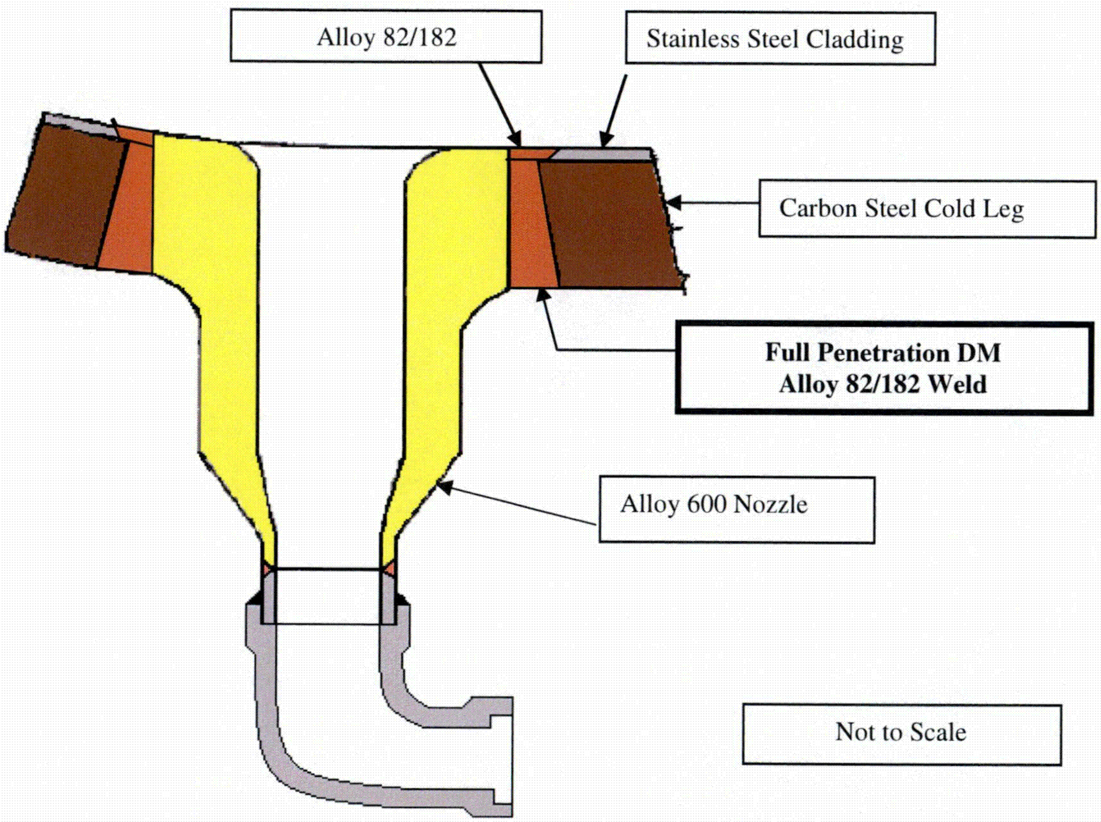
Grand Total 14564

(all dose stated in mrem)

ATTACHMENT 1

Figure 1

Nozzle Assembly Materials

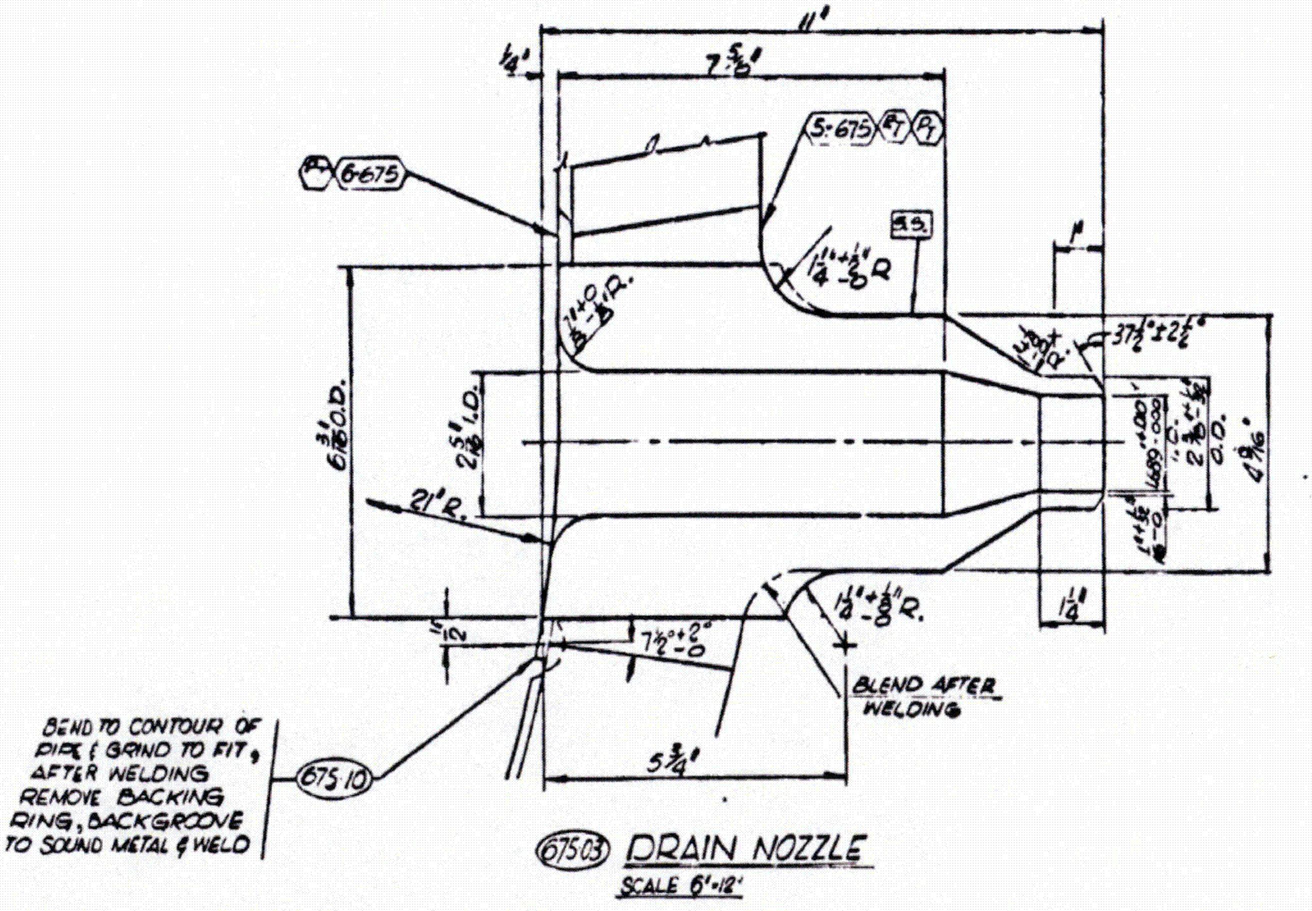


ATTACHMENT 1

Figure 2

Hot Leg Drain Nozzle Configuration (Representative)

(excerpt from PNP vendor drawing VEN-M1-D Sheet 108, Revision 8)



ATTACHMENT 2

The attached Structural Integrity Associates, Inc. report and supporting calculations provide the technical basis for the proposed alternative.

1. Structural Integrity Associates, Inc. Report No. 1400669.401.R0, "Evaluation of the Palisades Nuclear Plant Branch Line Nozzles for Primary Water Stress Corrosion Cracking," Revision 0, dated May 14, 2015.

2. Structural Integrity Associates, Inc. Calculation, File No. 1400669.313, "Crack Growth Analysis of the Hot Leg Drain Nozzle," Revision 0, dated May 11, 2015.

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7. Structural Integrity Associates, Inc. Calculation, File No. 1400669.322, "Cold Leg Bounding Nozzle Weld Residual Stress Analysis," Revision 0, dated May 5, 2015.

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