September 13, 2021

L-2021-184
10 CFR 50.55a

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

RE: St. Lucie Nuclear Plant, Unit 2
Docket No. 50-389
Renewed Facility Operating License NPF-16
Relief Request Number 19 - Request for an Alternative to ASME Code Case N-729-6 for Replacement Reactor Vessel Closure Head Penetration Nozzle 85

In accordance with the provisions of 10 CFR 50.55a(z)(2), Florida Power and Light (FPL) hereby requests Nuclear Regulatory Commission (NRC) approval of a proposed alternative to the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Code Case N-729-6, "Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds Section XI, Division 1," for use at the St. Lucie Plant.

On September 9, 2021 during visual examination, a relevant indication was noticed on the Unit 2 Reactor Vessel Closure Head (RVCH) surface adjacent to Control Element Drive Mechanism #85. The indication was a small thin translucent stain/film observed on the RVCH uphill of CEDM penetration #85 that ran down the head surface to and in contact with the penetration #85 annulus area. The plant was in a refueling outage when the observation was made.

To address this issue, FPL is requesting relief from Code Case N-729-6 paragraphs -3142.2 & -3200 for performing volumetric and/or surface exams of the St. Lucie Unit 2 RVCH penetration 85. As an alternative, FPL proposes to perform a bare metal visual examination (VE) of penetration 85 at the next refueling outage after cleaning and documentation of the as left condition (SL2-27 – Spring 2023). The proposed alternative and supporting information are presented in Attachment 1.

FPL requests approval of the proposed alternative by September 14, 2021.

There are no regulatory commitments contained in this submittal. If there are any questions or if additional information is required, please contact Mr. Wyatt Godes, St. Lucie Licensing Manager at (772) 467-7435.

Sincerely,

Wyatt Godes

Wyatt Godes
Licensing Manager
St. Lucie Nuclear Plant

WG/tf

Attachments:

1. 10 CFR 50.55a Request for an Alternative to ASME Code Case N-729-6 for Replacement Reactor Vessel Closure Head Penetration Nozzle 85

cc: USNRC Regional Administrator, Region II
USNRC Project Manager, St. Lucie Nuclear Plant, Units 1 and 2
USNRC Senior Resident Inspector, St. Lucie Nuclear Plant, Units 1 and 2

St. Lucie Unit 2
Fourth Inspection Interval
RELIEF REQUEST NUMBER 19 (RR19), Rev. 0

Proposed Alternative
In Accordance with 10 CFR 50.55a(z)(2)
Request for an Alternative to ASME Code Case N-729-6
for Replacement Reactor Vessel Closure Head Penetration Nozzle 85

1. ASME CODE COMPONENT(S) AFFECTED:

Component:	Replacement Reactor Vessel Closure Head (RVCH) nozzles
Code Class:	Class 1
Exam Category:	ASME Code Case N-729-6, *"Alternative Examination Requirements for PWR Reactor Vessel Upper Heads with Nozzles Having Pressure-Retaining Partial-Penetration Welds, Section XI, Division 1"*
Code Item No.:	B4.30
Description:	Control Rod Drive Mechanism (CEDM) Nozzles, Specifically Nozzle 85
Size:	4.000 Inch Outside Diameter
Materials:	RVCH – SA-508 Class 3 Nozzles – SB-167 N06690 (Alloy 690) Weld Material: ERNiCrFe-7, ENiCrFe-7 (Alloy 690 weld material)

There are 91 CEDM nozzles, 10 incore instrument (ICI) penetrations and 1 head vent welded to the inside surface of the RVCH with partial penetration J-groove welds

2. APPLICABLE CODE EDITION AND ADDENDA:

The Fourth Ten Year ISI interval Code of record for St. Lucie Unit 2 is the 2007 Edition with 2008 Addenda of ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." Examinations of the reactor vessel closure head (RVCH) penetrations are performed in accordance with ASME Code Case N-729-6 (Ref. 1) as conditioned by 10 CFR 50.55a(g)(6)(ii)(D) (2) through (8).

The manufacturing Code for St. Lucie Unit 2 RVCH: ASME Boiler and Pressure Vessel (BPV) Code, Section III, "Rules for Construction of Nuclear Power Plant Components, Division 1," 1989 Edition.

St. Lucie Unit 2
Fourth Inspection Interval
RELIEF REQUEST NUMBER 19 (RR19), Rev. 0

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3. APPLICABLE CODE REQUIREMENT:

The Code of Federal Regulations (CFR) 10 CFR 50.55a(g)(6)(ii)(D)(1), requires (in part):

> "*Augmented ISI requirements: Reactor vessel head inspections (1) Implementation.* Holders of operating licenses or combined licenses for pressurized-water reactors as of or after June 3, 2020 shall implement the requirements of ASME BPV Code Case N-729-6 instead of ASME BPV Code Case N-729-4, subject to the conditions specified in paragraphs (g)(6)(ii)(D)(2) through (8) of this section, by no later than one year after June 3, 2020. All previous NRC-approved alternatives from the requirements of paragraph (g)(6)(ii)(D) of this section remain valid."

Code Case N-729-6, Paragraph -3141, "Inservice Visual Examinations (VE)" states:

(a) The VE required by -2500 and performed in accordance with IWA-2200 and the additional requirements of this Case shall be evaluated by comparing the examination results with the acceptance standards specified in -3142.1.

(b) Acceptance of components for continued service shall be in accordance with -3142.

(c) Relevant conditions for the purposes of the VE shall include evidence of reactor coolant leakage, such as corrosion, boric acid deposits, and discoloration.

Code Case N-729-6, Paragraph 3142.1, "Acceptance by VE" states:

(a) A component whose VE confirms the absence of relevant conditions shall be acceptable for continued service.
(b) A component whose VE detects a relevant condition shall be unacceptable for continued service until the requirements of (1), (2), and (c) below are met.
  (1) Components with relevant conditions require further evaluation. This evaluation shall include determination of the source of the leakage and correction of the source of leakage in accordance with -3142.3.
  (2) All relevant conditions shall be evaluated to determine the extent, if any, of degradation. The boric acid crystals and residue shall be removed to the extent necessary to allow adequate examinations and evaluation of degradation, and a subsequent VE of the previously obscured surfaces shall be performed, prior to return to service, and again in the subsequent refueling outage. Any degradation detected shall be evaluated to determine if any corrosion has impacted the structural integrity of the component. Corrosion that has reduced component wall thickness below design limits shall be resolved through repair/replacement activity in accordance with IWA-4000.

St. Lucie Unit 2
Fourth Inspection Interval
RELIEF REQUEST NUMBER 19 (RR19), Rev. 0

(c) A nozzle whose VE indicates relevant conditions indicative of possible nozzle leakage shall be unacceptable for continued service unless it meets the requirements of -3142.2 or -3142.3.

Code Case N-729-6, Paragraph -3142.2, "Acceptance by Supplemental Examination" states:

A nozzle with relevant conditions indicative of possible nozzle leakage shall be acceptable for continued service if the results of supplemental examinations [-3200(b)] meet the requirements of -3130.

Code Case N-729-6, Paragraph -3142.3, "Acceptance by Corrective Measures or Repair/Replacement Activity" states:

A nozzle with relevant conditions indicative of possible nozzle leakage shall be acceptable for continued service if the results of supplemental examinations [-3200(b)] meet the requirements of -3130.

(a) A component with relevant conditions not indicative of possible nozzle leakage is acceptable for continued service if the source of the relevant condition is corrected by a repair/replacement activity or by corrective measures necessary to preclude degradation.
(b) A component with relevant conditions indicative of possible nozzle leakage shall be acceptable for continued service if a repair/replacement activity corrects the defect in accordance with IWA-4000.

Code Case N-729-6, Paragraph -3200, "Supplemental Examination" states:

(a) Volumetric or surface examinations that detect flaws which require evaluation in accordance with -3130 may be supplemented by other techniques to characterize the flaw (i.e., size, shape, and orientation).
(b) The supplemental examination performed to satisfy -3142.2 shall include volumetric examination of the nozzle tube and surface examination of the partial penetration weld, or surface examination of the nozzle tube inside surface, the partial penetration weld, and nozzle tube outside surface below the weld, in accordance with Figure 2, or the alternative examination area or volume shall be analyzed to be acceptable in accordance with Mandatory Appendix I. The supplemental examinations shall be used to determine the extent of the unacceptable conditions and the need for corrective measures, analytical evaluation, or repair/replacement activity.

St. Lucie Unit 2
Fourth Inspection Interval
RELIEF REQUEST NUMBER 19 (RR19), Rev. 0

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4. REASON FOR REQUEST:

Florida Power & Light Co. (FPL) is requesting relief from Code Case N-729-6 paragraphs -3142.2 & -3200 for performing supplemental volumetric and/or surface exams of the St. Lucie Unit 2 RVCH penetration 85. FPL performed a scheduled visual examination (VE) of the RVCH nozzle penetrations during the current St. Lucie Unit 2 refueling outage (SL2-26-Fall 2021) in accordance with Code Case N-729-6 (Ref. 1). This was the third visual examination (VE) of the replacement RVCH. During the examination, a relevant condition was noted on the RVCH surface adjacent to CEDM penetration 85 (See Figure 1 for relative location). The indication was a small thin translucent stain/film observed on the RVCH head uphill of CEDM penetration 85 and ran down the head surface to and in contact with the penetration 85 annulus area. The stain had no buildup/thickness on the surface of the RVCH or at the penetration 85 annulus and there was no evidence of rust or degradation of the carbon steel RVCH at the penetration 85 annulus. Although the stain was identified as a relevant indication, the appearance and pattern was not consistent with known operational boric acid leaks coming from RVCH penetrations as documented in EPRI Report MRP-60 (Ref. 2).

There was also evidence of a thin white film on top of the insulation running along a seam in the proximity of penetration 85; however, no trail from the top side of the insulation to the area of the stain on the RVCH could be identified. Efforts to positively identify the source of the thin film/stain at penetration 85 were inconclusive. Penetration 85 is adjacent to and uphill of ICI penetrations 98 & 99 which have mechanical connections that are disassembled every refueling outage and have had occurrences of reactor coolant system (RCS) spills in previous outages. However, the thin film on the RVCH could not be conclusively attributed to previous ICI disassembly and spillage of RCS.

As the stain on the head at penetration 85 was a thin film without accumulation, sufficient sample/scrapings could not be obtained for boric acid analysis. Wet swipe samples were obtained from the thin film in the area of interest at penetration 85, as well as areas away from the penetration 85 for comparison. The swipe samples were inconclusive on the determination of boric acid due to the lack of quantity on the stain. Isotopic analysis of the swipe samples were used to compare ratios of Co58:Co60 to that of reactor coolant samples prior to shutdown. Co58 has a half-life of ~70.86 days and Co60 has a half-life of ~1925.28 days. The ratio of Co58:Co60 in the shutdown reactor coolant was 2.90 and the Co58:Co60 ratio determined from the swipe sample taken from the penetration 85 area of interest was 0.04, which indicates that the thin film substance has been on the RVCH surface for a significant period of time and is not indicative of active pressure boundary leakage from the penetration 85 area. Other control samples away from penetration 85 had Co58:Co60 ratios of between 0.03 and 0.06 providing further evidence that the thin stain was not indicative of recent operational reactor coolant leakage.

Once a relevant condition of possible nozzle penetration leakage is identified, Code Case N-729-6, paragraph -3142.2 requires supplemental examination including a volumetric examination of the nozzle tube and surface examination of the partial penetration weld in accordance with paragraph -3200 (b) that demonstrates the acceptance of the nozzle.

In order to perform the Supplemental Examinations in accordance with Code Case N-729-6, paragraph -3200(b), it will be necessary to mobilize equipment and personnel to the site on an emergent basis. The RVCH head stand does not have an access door. This will require the RVCH to be lifted and placed in a temporary laydown area so that the polar crane can be used to remove sections of the head stand shield ring to gain access under the head. A temporary shield wall will need to be constructed and placement of the examination equipment inside the ring prior to placing the head back on the stand. This will require multiple heavy lifts of the RVCH to support the inspection. The additional work could delay the outage greater than a week depending on available resources. In addition, the supplemental examinations require access to the underside of the highly contaminated RVCH which would expose personnel to elevated dose rates. Dose rates under the RVCH are 3 R/hr based on a recent historical survey for Unit 2. The additional dose to modify the head stand, and perform the Supplemental Examination is estimated to be approximately 7.5 man-rem for this work.

Considering the appearance and analyzed properties of the film/stain are not indicative of reactor vessel head leakage, adding this extra duration to the outage and increasing the personnel dose for performing the Supplemental Examinations represents a hardship or unusual difficulty without a compensating increase in the level of quality and safety, pursuant to 10 CFR 50.55a(z)(2).

5. PROPOSED ALTERNATIVE AND BASIS FOR USE:

FPL is requesting relief from Code Case N-729-6 paragraphs -3142.2 & -3200 for performing volumetric and/or surface exams of the St. Lucie Unit 2 RVCH penetration 85. As an alternative, FPL proposes to perform a bare metal visual examination (VE) of penetration 85 at the next refueling outage after cleaning and documentation of the as left condition (SL2-27 – Spring 2023).

Previous Examinations of the St. Lucie Unit 2 Replacement Head

The St. Lucie Unit 2 replacement head went into service in January 2008 at St. Lucie Unit 2 (~14 years of service). Prior to installation, a preservice volumetric examination of the replacement RVCH nozzles was performed. There were no recordable indications identified during the preservice volumetric examinations of the nozzle tube in the area of the J-groove welds.

A bare metal visual examination (VE) was performed of the St. Lucie Unit 2 replacement RVCH in 2012 in accordance with ASME Code Case N-729-1 (Ref. 5), Table 1, Item B4.30. This visual examination was performed by VT-2 qualified examiners, on the outer surface of the RVCH including the annulus area of the penetration nozzles. This examination did not

St. Lucie Unit 2
Fourth Inspection Interval
RELIEF REQUEST NUMBER 19 (RR19), Rev. 0

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reveal any surface or nozzle penetration boric acid that would be indicative of nozzle leakage.

A bare metal visual examination (VE) was performed of the St. Lucie Unit 2 replacement RVCH in SL2-23, Spring 2017 in accordance with ASME Code Case N-729-1, Table 1, Item B4.30. This visual examination was performed by VT-2 qualified examiners on the outer surface of the RVCH including the annulus area of the penetration nozzles. This examination did reveal dried boric acid coming from multiple incore instrument (ICI) columns, including penetrations 98, 99 and 100, which are downhill but adjacent to penetration 85. The dried boric acid was removed and a follow up VE of the ICI locations was performed at the following refuel outage (SL2-24 – Fall 2018) of the area around the affected ICI penetrations, including penetrations 98, 99 and 100. No indication of penetration leakage was identified in the subsequent re-examination.

Note: All the above visual examinations (VE) were performed by qualified VT-2 examiners performing VE exams per Code Case N-729-1 and later versions have documented additional 4 hours of training in detection of borated water leakage per Ref. 1 and Ref. 5, Table 1 Note 2 as required by the FPL NDE procedure.

MRP-375 Information Regarding the Structural Adequacy & Performance of the RVCH Alloy 690 Nozzles

Evaluations were performed and documented in EPRI MRP-375 (Ref. 3) to demonstrate the acceptability of extending the RVCH inspection intervals for ASME Code Case N-729-1, item B4.40 components based on the superior laboratory and operational performance. Alloy 690 is highly resistant to Primary Water Stress Corrosion Cracking (PWSCC) due to its approximate 30% chromium content. Per MRP-115 (Ref. 4), it was noted that Alloy 82 crack growth rate (CGR) is 2.6 times slower than Alloy 182. There is no strong evidence for a difference in Alloy 52 and 152 CGRs. Therefore, data used to develop factors of improvement for Alloy 52/152 were referenced against the base case Alloy 182, as Alloy 182 is more susceptible to initiation and growth when compared to Alloy 82. A simple factor of improvement (FOI) approach was applied in a conservative manner in MRP-375 using multiple data. Based on plant service experience, FOI studies using laboratory data, deterministic study results, and probabilistic study results, MRP-375 documented the basis for extended inspection intervals. This information documents the structural suitability of the RVCH for extended periods of time.

Per MRP-375, much of the laboratory data indicated an FOI of 100 for Alloy 690/52/152 versus Alloy 600/182/82 (for equivalent temperature and stress conditions) in terms of crack growth rates (CGR). In addition, laboratory and plant data demonstrate an FOI in excess of 20 in terms of the time to PWSCC initiation. This reduced susceptibility to PWSCC initiation and growth supports elimination of all volumetric examinations throughout the plant service period, and by extension, supports not performing Supplemental Examinations this refueling outage.

Deterministic calculations demonstrate that the alternative volumetric re-examination schedule of MRP-375 (Table 4-1) of every 20 years is sufficient to detect any PWSCC before it could develop into a safety significant circumferential flaw that approaches the large size (i.e., more than 300 degrees of circumferential extent) necessary to produce a nozzle ejection with significant margins of safety. The deterministic calculations also demonstrate that any base metal PWSCC would likely be detected prior to a through-wall flaw occurring. Probabilistic calculations based on a Monte Carlo simulation model of the PWSCC process, including PWSCC initiation, crack growth, and flaw detection via ultrasonic testing, show a substantially reduced effect on nuclear safety compared to a RVCH with Alloy 600 nozzles examined per current requirements.

Service Experience

As documented in MRP-375 (published in 2014), the resistance of Alloy 690 and corresponding weld metals Alloy 52 and 152 is demonstrated by the lack of any PWSCC indications reported in these materials, in up to 24 calendar years of service for thousands of Alloy 690 steam generator tubes, and more than 22 calendar years of service for thick-wall and thin-wall Alloy 690 applications. There has been no new operating experience of PWSCC identified since the publication. This excellent operating experience includes service at pressurizer and hot-leg temperatures and includes Alloy 690 wrought base metal and Alloy 52/152 weld metal. This experience includes ISI volumetric or surface examinations performed in accordance with ASME Code Case N-729-1 on at least 13 of the 41 replacement RVCHs currently operating in the U.S. fleet. This data supports a factor of improvement in time of at least 5 to 20 to detectable PWSCC when compared to service experience of Alloy 600 in similar applications.

Two of the replacement heads that were volumetrically examined in accordance with N-729-1 were Turkey Point Units 3 and 4, owned by FPL. The Turkey Point heads were replaced in 2004 and 2005 respectively and examined during their 2014 refueling outages. The St. Lucie Unit 2 head and the Turkey Point Units 3 & 4 heads were fabricated by the same manufacturer (Areva), using thermally treated Alloy 690 nozzle material produced by the same material supplier (Valinox Nucleaire), per the same ASME SB-167 nozzle material specifications with identical supplemental requirements as the previously examined Turkey Point Units 3 and 4 heads. The nozzle J-groove attachment welds for the Turkey Point and St. Lucie Heads utilized PWSCC resistant ERNiCrFe-7 (UNS N06052 and/or ENiCrFe-7 UNS W86152) weld materials.   The St. Lucie Unit 2 and Turkey Point Units 3 & 4 were all procured to ASME Section III, 1989 Edition, no addenda. Areva also fabricated the St. Lucie Unit 1 replacement pressurizer that contains 120 alloy 690 (UNS N06690) pressurizer heater sleeves that utilized PWSCC resistant ERNiCrFe-7 (UNS N06052 and/or ENiCrFe-7 UNS W86152) weld materials. The St. Lucie Unit 1 replacement pressurizer was installed in 2005 and operates at 653°F, approximately 50°F higher than the St. Lucie Unit 2 RVCH, with no indication of PWSCC degradation.

Design Features Further Increasing the Resistance of the St. Lucie Unit 2 Replacement Head to PWSCC

In addition to the standard Alloy 690 materials (plate and CRDM nozzle material) test data reported in MRP-375, FPL imposed supplemental requirements on the St. Lucie Unit 2 nozzle materials (identical to and the previously examined Turkey Point Unit 3 & 4 heads) to increase the material resistance to PWSCC. These supplemental requirements include; thermal treatment (TT), prohibition of cold straightening after TT, ingot remelting to reduce impurities, additional chemistry requirements, microstructure and grain size requirements. These methods substantially reduce PWSCC susceptibility beyond that assumed in the generic MRP-375 study, resulting in additional assurance that the St. Lucie Unit 2 head penetrations are highly resistant to PWSCC.

As stated above, none of the prior examinations of replacement RVCHs and pressurizer with Alloy 690 nozzles have revealed any indications of PWSCC or service-induced cracking.

Enhance RCS Leakage Detection at St. Lucie Unit 2 Provides Defense in Depth

As discussed above, the initiation or growth of a safety significant flaw in an alloy 690 base material and associated weld material in a RVCH penetration is extremely unlikely. However, as an added measure of safety, the industry imposed an NEI-03-08 “needed” requirement, to improve their RCS leak detection capability in part due to the concern with PWSCC or alloy 600 materials. St. Lucie Unit 2 has adopted the standardized approach to measuring RCS leak rate in WCAP-16423 (Ref.6) and has incorporated the action levels in WCAP-16465 (Ref. 7). The enhanced leak rate monitoring and detection procedure monitors specific values of unidentified leakage, seven day rolling average, and baseline means. Action levels are initiated as low as when the unidentified leak rate exceeds 0.1 gpm. The enhanced leak detection capability provides an increased level of safety that if a flaw were to grow through wall, although unlikely, that is would be detected prior to it growing to a safety significant size.

Conclusion

FPL has concluded that there is reasonable assurance that the relevant indication at the St. Lucie Unit 2 RVCH penetration 85 is not indicative of RCS leakage from penetration 85 base material or partial penetration weld based on the following:

- The relevant indication has the appearance of a thin film/stain with no thickness, which is not the characteristic of an active RCS leak as documented in MRP-60 (Ref. 2)

- Isotopic chemistry analysis of swipe samples of the thin film/stain at penetration 85 using the ratio of Co58:Co60 indicates the stain was not indicative of active pressure boundary leakage from the penetration 85 area.
- There was no rust discoloration or degradation of the carbon steel head surface or in the penetration annulus area.
- Although there was some evidence of a similar thin film on the insulation above/near penetration 85, the source of the thin film could not be conclusively identified.

In addition, the St. Lucie Unit 2 RVCH has only been in service since 2008 with less than 14 years of operation. Operating experience and laboratory testing of Alloy 690 materials and the associated alloy 52/152 weld materials (ERNiCrFe-7 and/or ENiCrFe-7) show significant resistance to PWSCC with factors of improvement over alloy 600 materials and supports reinspection intervals of 20 years with margins of safety.

As discussed above, performing the Supplemental Examinations in accordance with Code Case N-729-6, paragraph -3200(b), will result in additional work activities that could delay the outage greater than a week depending on available resources and subject worker to approximately 7.5 man Rem of dose.

Based on the discussion and the summary above, it is requested that the NRC authorize this proposed alternative in accordance with 10 CFR 50.55a(z)(2) as the alternative provides an acceptable level of quality and safety

6. DURATION OF PROPOSED ALTERNATIVE:

The proposed alternative is for one refueling outage, to the end of operating cycle 26 (Spring 2023), at which time a bare metal visual examination (VE) of penetration 85 meeting the acceptance requirements in Code Case N-729-6 will be performed.

7. PRECEDENTS:

1) NRC letter regarding approval of Relief Request (RR) 14 for Fort Calhoun Station, Unit No. 1, Subject: Fort Calhoun Station, Unit No. 1 - Request for Relief RR-14, From Certain Requirements of ASME Code Case N-729-1 for Reactor Vessel Head Penetration Nozzle Welds, dated August 21, 2015 (ADAMS Accession number ML15232A003)

2) NRC letter regarding approval of Relief Request 57 for Palo Verde Generating Station, Unit 1 - Relief Request No. 57 To Approve Alternate Requirements For The Reactor Pressure Vessel Head Nozzles To Perform A Bare Metal Examination Per ASME Code Case N-729-4, dated February 20, 2018 (ADAMS Accession number ML18040A331)

St. Lucie Unit 2
Fourth Inspection Interval
RELIEF REQUEST NUMBER 19 (RR19), Rev. 0

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8. REFERENCES:

1. ASME Code Case N-729-6, "Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds, Section XI, Division 1," Approved March 3, 2016.
2. *Materials Reliability Program: Visual Examination for Leakage of PWR Reactor Vessel Upper Head Nozzles:(MRP-60, Rev 5).* EPRI, Palo Alto, CA: 2018. 3002013268 [ML020090G363 – Transmittal of Proprietary MRP-60, Rev 5]
3. *Materials Reliability Program: Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles (MRP-375)*, EPRI, Palo Alto, CA: 2014. 3002002441. [freely available at www.epri.com]
4. *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. [freely available at www.epri.com]
5. ASME Code Case N-729-1, "Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds, Section XI, Division 1," Approved March 28, 2006.
6. WCAP-16423-NP, Rev. 0, "Pressurized Water Reactor Owners Group Standard Process and Methods for Calculating RCS Leak Rate for Pressurized Water Reactors," Westinghouse Electric Co., September 2006. (Transmitted to the NRC – ML070310081)
7. 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. (Transmitted to the NRC – ML070310081)

St. Lucie Unit 2
Fourth Inspection Interval
RELIEF REQUEST NUMBER 19 (RR19), Rev. 0

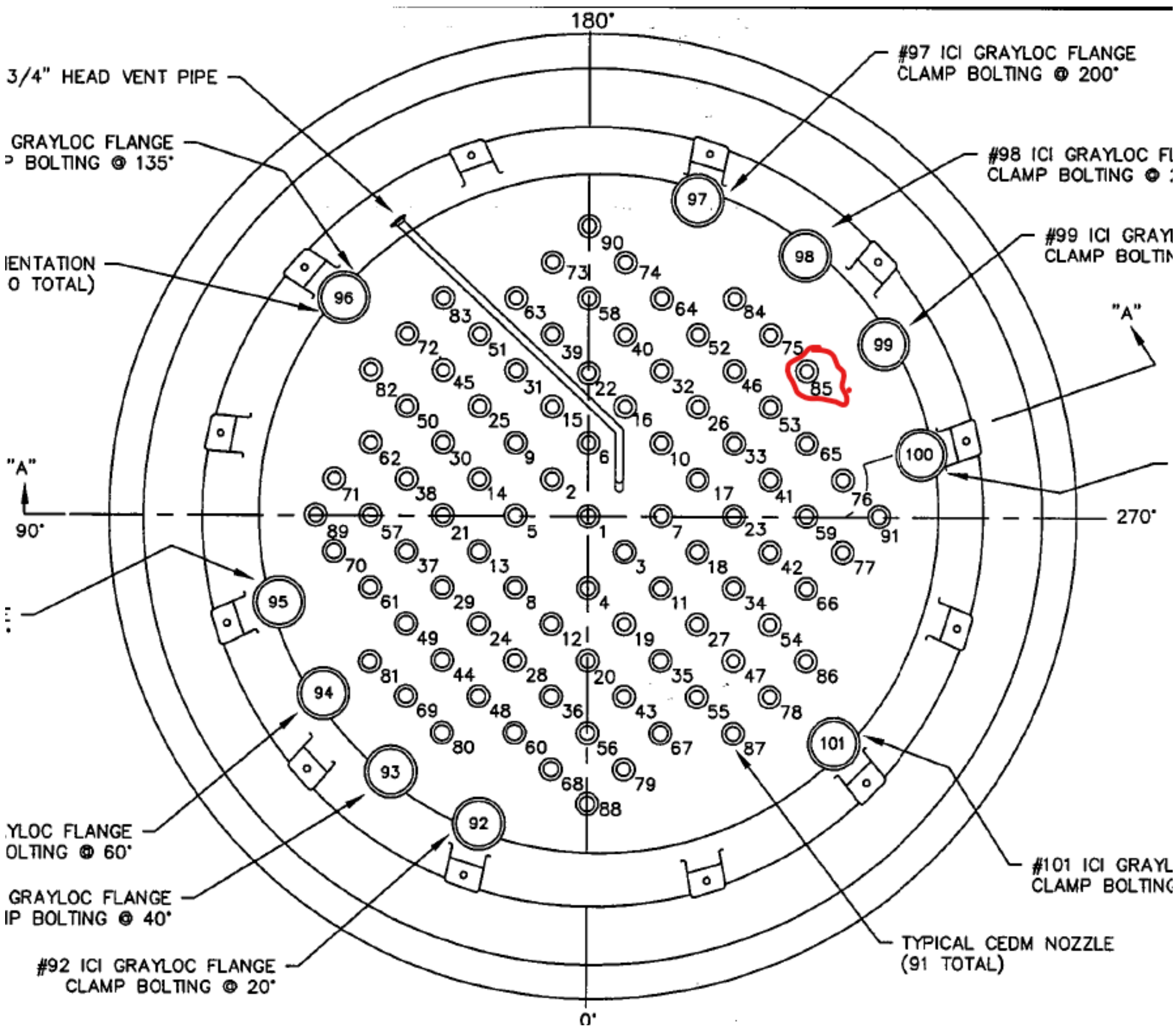

Figure 1: St. Lucie Unit 2 RVCH Penetration Layout with Penetration 85 Identified