NEXTera ENERGY SEABROOK

September 26, 2011

SBK-L-11191
Docket No. 50-443

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

Seabrook Station

Third Ten-Year Interval Inservice Inspection Relief Requests

Pursuant to 10 CFR 50.55a(a)(3)(i), 10 CFR 50.55a(a)(3)(ii) and 10 CFR 50.55a(g)(5)(iii), NextEra Energy Seabrook, LLC (NextEra) requests NRC approval for relief requests applicable to the Third Ten-Year Interval Inservice Inspection Program. Attachments 1 through 3 contain relief requests 3IR-1, 3IR-2, and 3IR-3 respectively.

Attachments 1 and 2 are relief requests documenting examinations that are impractical. Attachment 3 is a relief request proposing use of an alternative to a code requirement. The relief requests were submitted and approved during the previous ten-year (10-Yr) Inservice Inspection (ISI) interval. Since the access or configurations have not changed, relief is also required for the 3rd 10-Yr. ISI interval. Relief request 3IR-1 requests relief from ASME Code requirements, based on impracticality due to component design. Relief request 3IR-2 requests relief from ASME Code requirements, due to hardship or unusual difficulty accessing components. Relief request 3IR-3 requests approval to use an alternative examination method than specified in the ASME Code.

Relief requests 3IR-1 through 3IR-3 have been formatted in accordance with NEI White Paper Rev. 1, "Standard Format for Requests from Commercial Reactor Licensees Pursuant to 10 CFR 50.55a."

NextEra respectfully requests approval of these requests by September 30, 2012, in order to prepare for inspections during the fall 2012 refueling outage.

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NRR

If you have any questions regarding this submittal, please contact Mr. Michael O'Keefe, Licensing Manager at (603) 773-7745.

Sincerely,

NextEra Energy Seabrook, LLC

Paul O. Freeman
Site Vice President

Attachments:

cc:	W.M. Dean,	NRC Region I Administrator
	G. E. Miller,	NRC Project Manager, Project Directorate I-2
	W. J. Raymond,	NRC Resident Inspector

Attachment 1

Relief Request 3IR-1

Examination Category B-B
Pressure Retaining Welds in Vessels Other Than Reactor Vessels

NextEra Energy Seabrook, LLC
Third Ten-Year Interval
10 CFR 50.55a Request Number 3IR-1, Rev. 0

Relief Request
in Accordance with 10 CFR 50.55a(g)(5)(iii)

--Inservice Inspection Impracticality--

Request for Relief for Steam Generator Main Steam Outlet Nozzle Inside Radius Section

1. ASME Code Components Affected

Code Class:	2
System:	RC
Examination Categories:	C-B, Pressure Retaining Nozzle Welds in Vessels
ISI Component ID:	RC E-11A 16-IR

2. Applicable Code Edition and Addenda

NextEra Energy Seabrook, LLC (NextEra) is currently in the 3rd 10-Year Inservice Inspection (ISI) Interval. The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) of record for the current 10-Year ISI interval is Section XI, 2004 Edition.

3. Applicable Code Requirements:

ASME Section XI, 2004 Edition, Table IWC-2500-1

Category C-B, Pressure Retaining Nozzle Welds in Vessels

Item No. C2.22, Nozzle Inside Radius Section

ASME Section XI, 2004, Table IWC-2500-1 Category C-B, Item No. C2.22 – Nozzle Inside Radius Section requires that the inner radius sections of all nozzles at terminal ends of piping runs be volumetrically examined.

Note 4 of Table IWC-2500-1, Category C-B states "in the case of multiple vessels of similar design, size, and service (such as steam generators, heat exchangers), the required examinations may be limited to one vessel or distributed among the vessels."

4. Impracticality of Compliance

Pursuant to 10CFR50.55a(g)(5)(iii), NextEra has determined that due to design and geometry, the volumetric examination requirement for nozzle inside radius section of the

Steam Generator Main Steam Outlet Nozzle Inner Radius, RC E-11A 16-IR as specified in Table IWC-2500-1, Examination Category C-B, Item No. C2.22 is impractical to meet.

The steam generator main steam outlet nozzle is one piece containing a set of seven holes bored parallel to the nozzle centerline. These seven flow limiting bores make a square transition (no inner radius) to the nozzle making it ultrasonically impractical to examine. In addition, this nozzle design does not match typical figures in Figure IWC-2500-4.

Limitation sketch is provided in Figure 3IR-1-1.

5. Burden Caused by Compliance

To perform an inner radius examination, the main steam outlet nozzle would require modification/replacement. This type of modification/replacement would be impractical and would not provide an increase in quality and safety.

6. Proposed Alternative And Basis for Use

The geometry of this nozzle design, with the bored flow restrictor holes, does not result in an actual inner radius, and therefore, no meaningful examination can be performed. This design does not entail a nozzle with a radius as described in Figure IWC-2500-4, but instead has several "corners," corresponding to each bored hole. As a result, the design of the nozzle is not applicable to the Code requirement and compliance with the Code should not be required. Therefore, no alternate examinations of inner radius section RC E-11A 16-IR are proposed.

A VT-2 examination associated with the system pressure test is performed on this nozzle each inspection period as specified in Table IWC-2500-1, Examination Category C-H of the 2004 Edition of ASME Section XI. The required VT-2 visual examination provides reasonable assurance of continued structural integrity.

7. Duration of Proposed Alternative

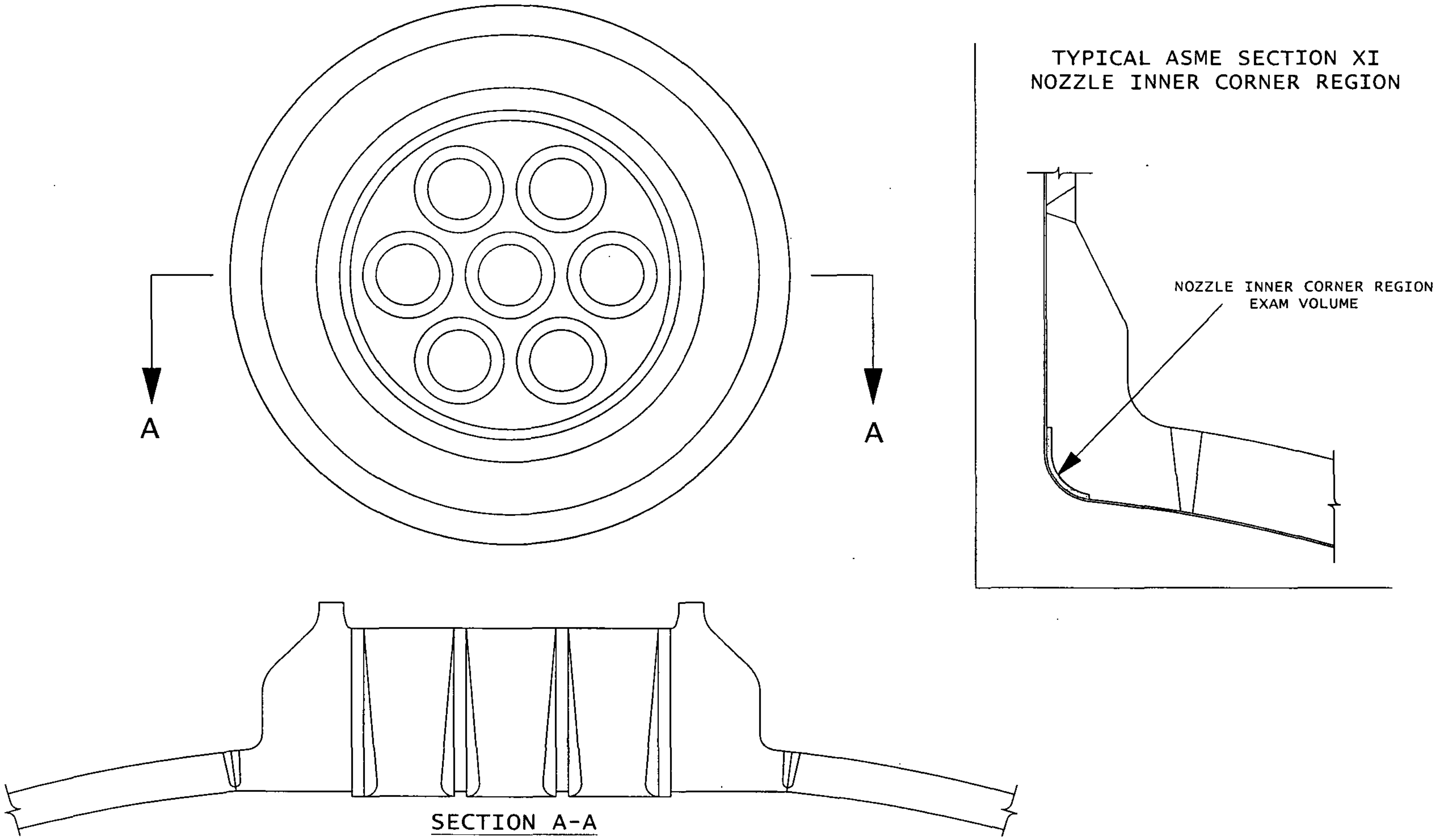
The alternative requirements of this request will be applied for the remaining duration of the current 3rd 10-year ISI interval.

8. Precedents

- Second interval relief request 2IR-4 Rev. 0 was approved for Seabrook Station by the NRC in a Safety Evaluation Report dated March 21, 2001 (TAC No. MA9902) (ML010540162).

3IR-1, Rev. 0 Figure 3IR-1-1

SEABROOK SG MAIN STEAM OUTLET NOZZLE GENERIC DETAILS



(SHOWING ABSENCE OF INNER CORNER REGION AT RC E-11A 16-IR)

Attachment 2

Relief Request 3IR-2

Examination Category B-D
Full Penetration Welded Nozzles in Vessels

Proposed Alternative
in Accordance with 10 CFR 50.55a(a)(3)(ii)

--Hardship or Unusual Difficulty
Without Compensating Increase in Level of Quality or Safety--

Request for Relief for Pressurizer Supports

1. ASME Code Components Affected

Code Class:	1
System:	RC
Examination Category:	F-A
Item No:	F1.40, Supports Other Than Piping
ISI Component ID:	RC E-10 A-LUG Support RC E-10 B-LUG Support RC E-10 C-LUG Support RC E-10 D-LUG Support

2. Applicable Code Edition and Addenda

NextEra Energy Seabrook, LLC (NextEra) is currently in the 3rd 10-Year Inservice Inspection (ISI) Interval. The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) of record for the current 10-Year ISI Interval is Section XI, 2004 Edition.

3. Applicable Code Requirement

ASME Section XI, 2004 Edition, Table IWF-2500-1

Category F-A, Supports

Item No. F1.40, Supports Other Than Piping

4. Reason for Request

Pursuant to 10CFR50.55a(a)(3)(ii), relief is requested from performing the VT-3 visual examination of the four Pressurizer supports on the basis that meeting the Code requirement presents unusual difficulty.

A 15" thick concrete shield wall weighing approximately 85,000 pounds surrounds the NextEra Pressurizer approximately three quarters of the way around. The clearance between the shield wall and the Pressurizer vessel with insulation is approximately 12", with less clearance at the top cubicle opening due to structural steel. The north end of the cubicle has greater vessel to shield wall clearance, but this is where safety valve piping and spray piping run. Ladders or platforms do not exist to make the examination area accessible nor can any ladders be placed due to restrictions by piping, conduit and other attachments.

The pressurizer lugs are located on the pressurizer at elevation 23'-6". Potential access is gained from either above the lugs or from below. Potential access from above is gained by climbing a ladder on the outside of the shield wall at elevation 25' and entering the cubicle at the top of the pressurizer at elevation 50'. At the top of the pressurizer, safety valve structural steel is used for footing as no platform exists in the cubicle. Access from the top must be made from the north side of the cubicle where the pressurizer to shield wall distance is greatest (see Section A-A of Figure 3IR-2-2). From this location it is approximately 26'-6" to the lug elevation. There is no installed ladder within the pressurizer cubicle to allow for normal access and egress to the lug elevation from the top (see Figure 3IR-2-2). The elevation distance, amount of obstructions and attachments, and insulation renders remote visual equipment unusable. From below, lug access is not achievable due to a permanent ventilation duct that encircles the pressurizer (See Figure 3IR-2-1).

5. Proposed Alternative and Basis for Use:

No alternate examinations for the Pressurizer supports are proposed.

The unusually difficult normal and emergency access/egress needed inside this highly restricted enclosure to remove insulation to perform the VT-3 visual examinations would result in unusual difficulty without a compensating increase in quality and safety.

A likely failure mechanism of these supports would involve a transient or seismic activity, which could impose rotational forces on the Pressurizer. Attached lugs that exist between these supports could impart forces on the supports from a transient or seismic event. There has been no documented seismic event or transient affecting the Pressurizer. Therefore, the most probable failure mechanism that could occur to the subject supports would be corrosion of the supports. Visual examinations (VT-3) of other accessible components within the Pressurizer cubicle have shown no evidence of corrosion.

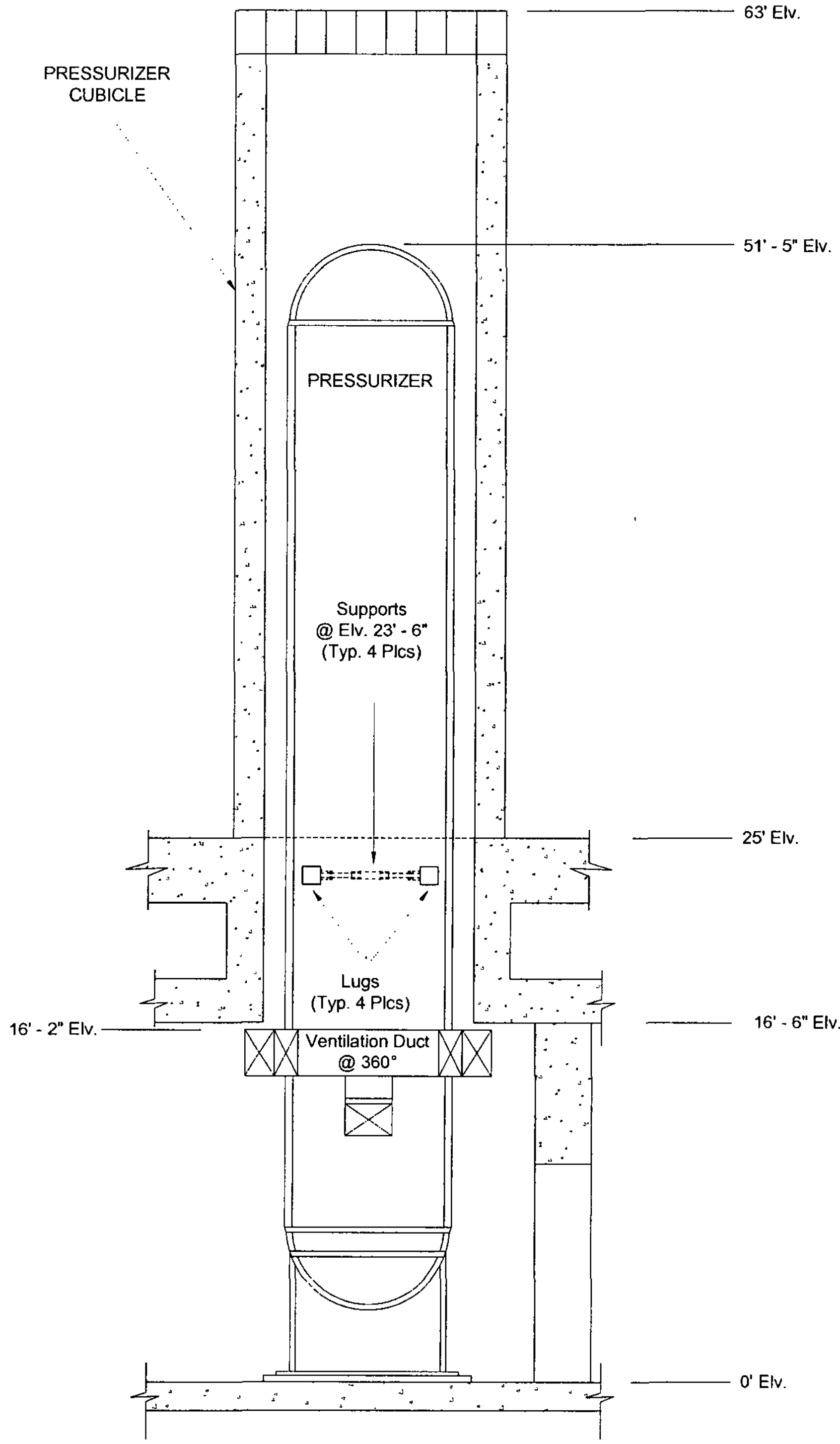
These supports are subject to VT-2 visual examination as part of the system leakage test on the Pressurizer vessel conducted each refueling outage as specified in Table IWB-2500-1, Examination Category B-P of the 2004 Edition of ASME Section XI. As part of the visual examination, VT-2 examiners physically enter the elevation just below the Pressurizer ventilation ductwork (0'), and observe the area for evidence of leakage, corrosion and boric acid that may be indicative of corrosion and wear of the subject supports. Based on acceptable results of the VT-2 visual examinations performed during system leakage tests, there is reasonable assurance of continued structural integrity of the subject supports.

6. Duration of Proposed Alternative

The alternative requirements of this request will be applied for the remaining duration of the current 3rd 10-year ISI interval.

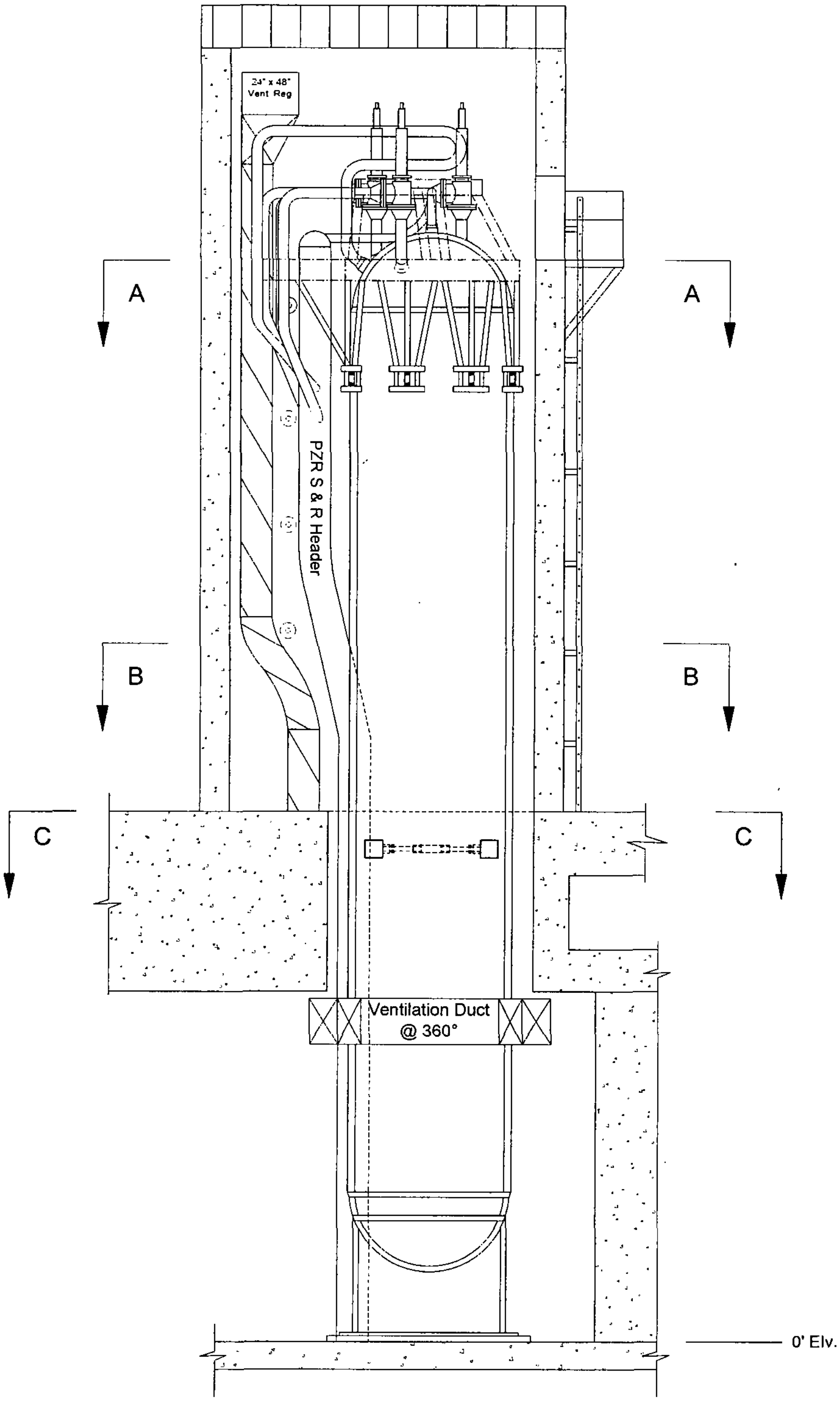
7. Precedents

- A similar first interval relief request, IR-12 Rev. 0 was approved for Seabrook Station by the NRC in a letter dated September 3, 2002 (TAC No. MB2561)(ML021990725)
- A similar second interval relief request, 2IR-12 Rev. 1 was approved for Seabrook Station by the NRC in a letter dated July 20, 2009 (TAC No. MD9781) (ML091830415)

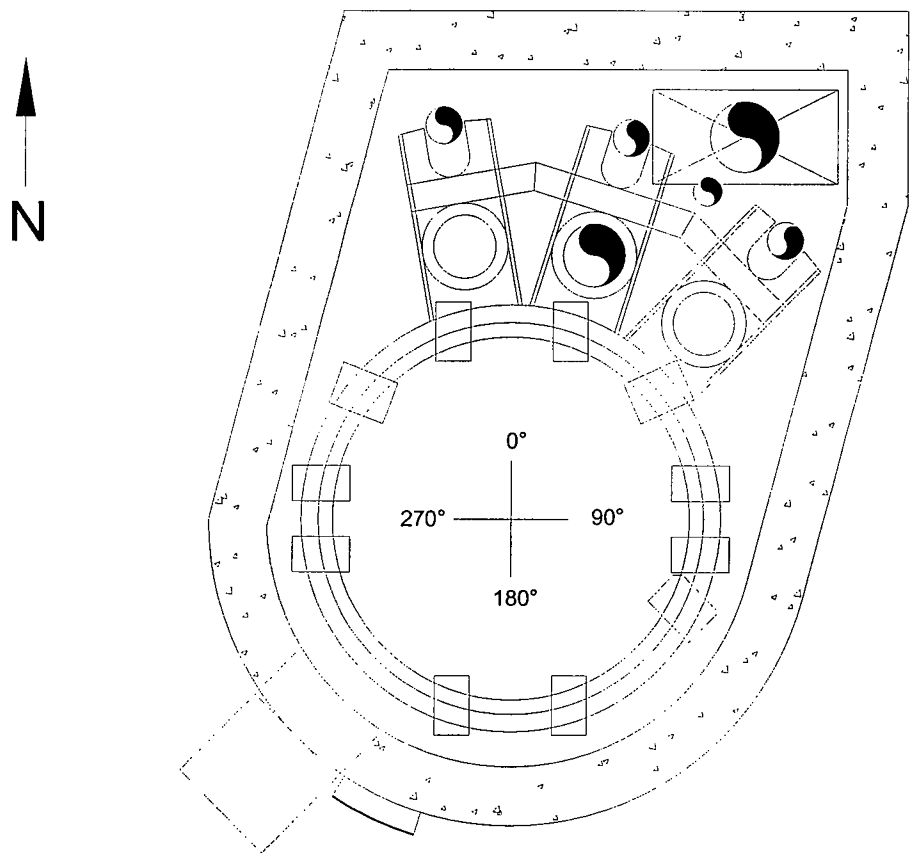


VIEW LOOKING NORTH

(GENERIC DETAILS)

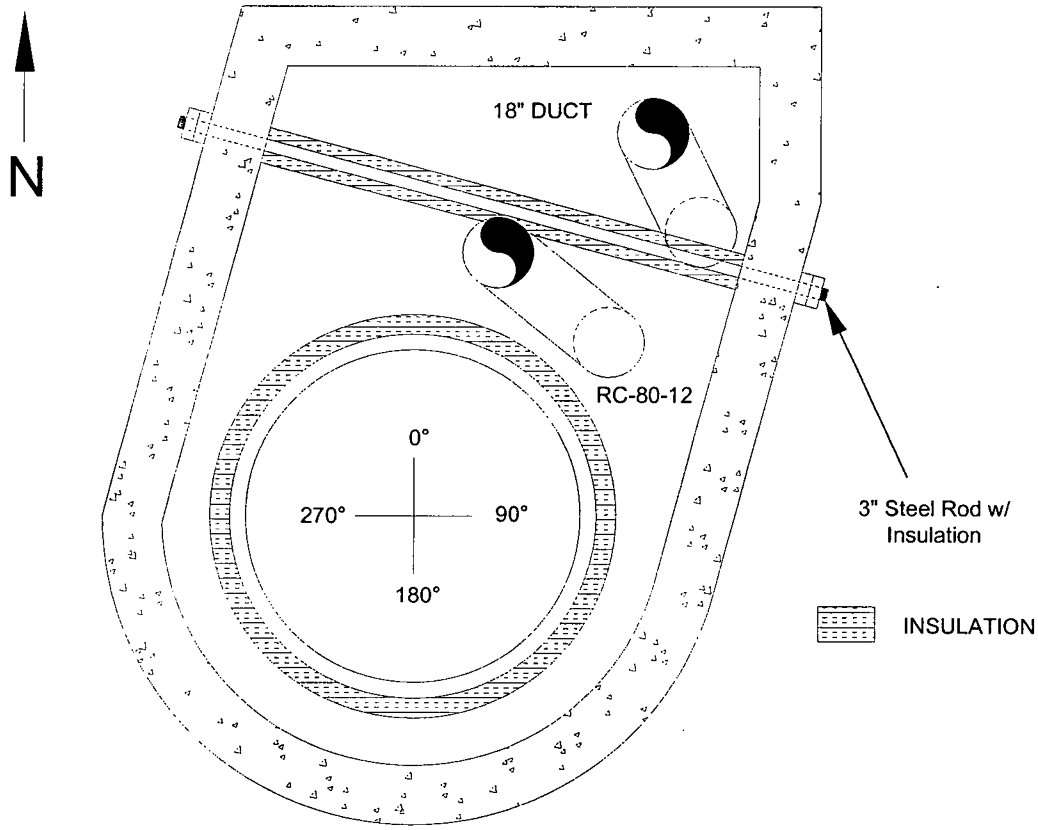


VIEW LOOKING EAST
(DETAILED VIEW)



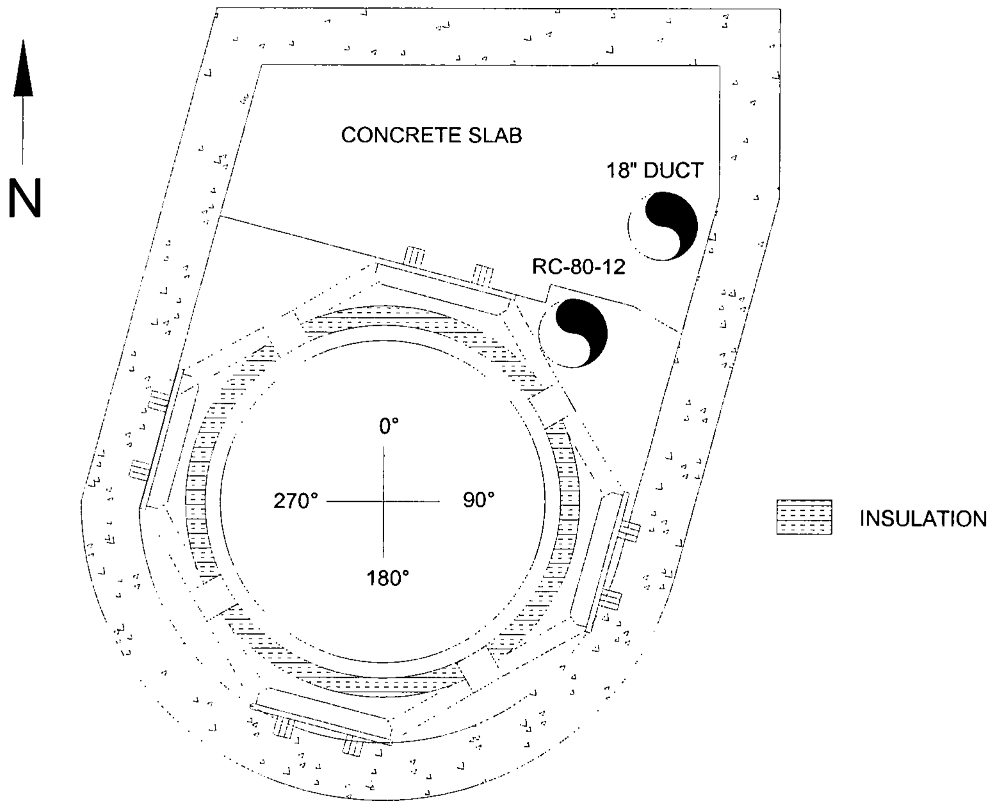
SECTION A-A

(Elv. 51')



SECTION B-B

(Elv. 32')



SECTION C-C

(Elv. 25')

Attachment 3

Relief Request 3IR-3

Examination Category C-B
Pressure Retaining Nozzle Welds in Vessels

NextEra Energy Seabrook, LLC
Third Ten-Year Interval
10 CFR 50.55a Request Number 3IR-3, Rev. 0

Proposed Alternative
in Accordance with 10 CFR 50.55a(a)(3)(i)

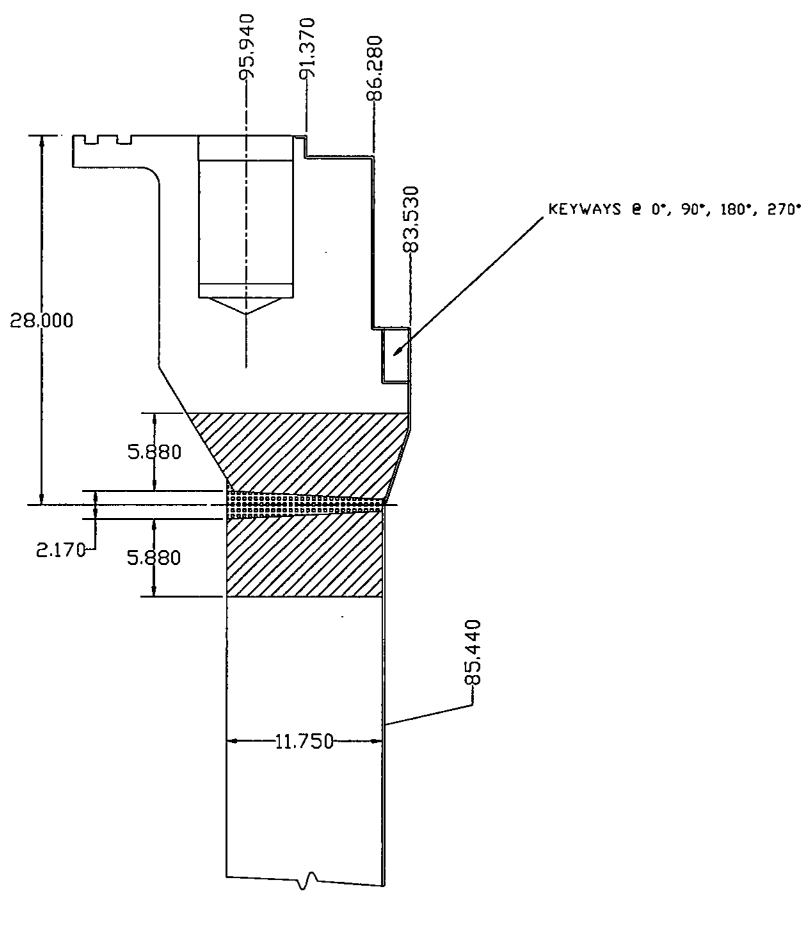
--Alternative Provides Acceptable Level of Quality or Safety--

Request for Relief to use PDI Demonstrated Ultrasonic Techniques for the Examination of the Reactor Pressure Vessel Flange-to-Upper Shell Weld

1. ASME Code Components Affected

Code Class:	1
System:	RC
Examination Categories:	B-A
Item No.:	B1.30, Shell-to-Flange Weld
ISI Component ID:	RC RPV-101-121

2. Component Detail Drawing:



3. Applicable Code Edition and Addenda

NextEra Energy Seabrook, LLC (NextEra) is currently in the 3rd 10-year Inservice Inspection (ISI) interval. The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) of record for the current 10-year ISI interval is Section XI, 2004 Edition (Reference 1).

4. Applicable Code Requirement

ASME Section XI, 2004 Edition, Appendix I, Article I-2100, paragraph (b) requires "Ultrasonic examination of reactor vessel-to-flange welds shall be conducted in accordance with Article 4 of ASME Section V, except that alternative examination beam angles may be used. These examinations shall be further supplemented by Table I-2000-1."

5. Reason for Request

NextEra is required to perform volumetric examination of all Reactor Pressure Vessel (RPV) welds during the third ten-year ISI interval pursuant to 10CFR50.55a. The Code requires that Ultrasonic (UT) examination of RPV welds, excluding the vessel-to-flange weld, shall be with techniques that have been demonstrated in accordance with AMSE Code Section XI, Appendix VIII. Further, in accordance with Appendix I, Paragraph I-2110(b), "Ultrasonic examination of reactor vessel-to-flange welds, closure head-to-flange welds, and integral attachment welds shall be conducted in accordance with Article 4 of Section V, except that alternative examination beam angles may be used."

Examination from the inside surface provides the best access for examination of the RPV shell-to-flange weld. The flange forging contains both inside and outside surface tapers, the outside taper angle is more than twice the angle of the inside surface taper. While both tapers will interfere with the examination to some degree, the inside surface taper provides the least amount of interference. Additionally, the outside surface of the RPV is typically inaccessible due to its placement inside the biological-shield wall and the installed insulation. Examination of this weld from the outside surface would require the removal of the installed insulation and access beneath the cavity seal ring. These efforts would result in significant personnel radiation exposure without a compensating increase in the level of quality and safety.

Although the reactor vessel-to-flange weld is specifically excluded from the referenced codes requiring Appendix VIII/PDI qualified techniques, NextEra believes that performing the reactor vessel-to-flange weld examination with PDI qualified personnel and procedures from the inside surface will provide an acceptable level of quality and safety.

6. Proposed Alternative And Basis for Use

In lieu of requirements specified in the ASME Code, Section XI, Appendix I, Subarticle I-2110, Paragraph (b), NextEra proposes to use procedures, personnel, and equipment qualified to the requirements of ASME Section XI Appendix VIII, Supplements 4 and 6 of the 2004 Edition, as administered by the Electric Power Research Institute's (EPRI) PDI program to conduct the vessel-to-flange weld examination. The RPV examination vendor will perform examinations designed to achieve the maximum coverage possible utilizing PDI qualified procedures and personnel from the inside surface. The proposed alternative represents the best techniques, procedures, and qualifications available to perform UT examinations of RPV welds. The PDI program addresses qualification requirements for each of the supplements that are defined in Appendix VIII of ASME Section XI.

The listed weld is the only circumferential shell weld in the RPV that is not examined with ASME Section XI, Appendix VIII techniques, as mandated in 10 CFR 50.55a. This rule mandates the use of ASME Section XI, Appendix VIII, Supplements 4 and 6 for the conduct of all other RPV weld examinations. Per Appendix I, Article I-2100, paragraph (b), ASME Section V, Article 4 techniques shall be used for the listed weld. ASME Section V, Article 4 describes generic examination techniques to be used for UT of welds. The calibration techniques, recording criteria and flaw sizing methods are based upon the use of a distance-amplitude-correction curve (DAC) derived from machined reflectors in a basic calibration block. UT performed in accordance with Section V, Article 4, uses recording thresholds known as percent of DAC for recording and reporting of indications within the examination volume. Indications detected in the exam volume, with amplitudes below these thresholds, are not required to be recorded and/or evaluated. Use of the Appendix VIII qualified techniques would enhance the quality of the examination.

The detection criterion is more conservative and the procedure requires the examiner to evaluate all indications determined to be flaws regardless of their amplitude. The recording thresholds in Section V, Article 4 are generic and do not take into consideration such factors as flaw orientation, which can influence the amplitude of UT responses.

EPRI Report NP-6273, "Accuracy of Ultrasonic Flaw Sizing Techniques for Reactor Pressure Vessels," dated March 1989, contains a comparative analysis of sizing accuracy for several different techniques. The results show that UT flaw sizing techniques based on tip diffraction are the most accurate. The proposed alternative Appendix VIII UT qualified detection and sizing methodologies use analysis tools based upon echo dynamics and tip diffraction. This methodology is considered more sensitive and accurate than the Section V, Article 4 processes. Procedures, equipment and personnel qualified via the PDI Appendix VIII, Supplement 4 and 6 programs have been demonstrated to have a high probability of detection and are generally considered superior to the techniques employed during earlier Section V, Article 4 RPV examinations. Accordingly, approval of this alternative examination and evaluation process is requested pursuant to 10 CFR 50.55a (a)(3)(i).

7. Duration of Proposed Alternative

The alternative requirements of this request will be applied for the remaining duration of the current 3rd 10-year ISI interval.

8. Precedents

Similar relief requests have been granted to the following plants:

- Second interval relief request 2IR-15 Rev. 0 was approved for Seabrook Station by the NRC in a Safety Evaluation Report dated April 7, 2009 (TAC No. MD9784) (ML090690557)

- NRC Safety Evaluation dated October 20, 2004, for Catawba Nuclear Station, Units 1 and 2; McGuire Nuclear Station, Unit 2, and Oconee Nuclear Station, Unit 3, dated July 14, 2004, "Request for Relief for Use of an Alternate to the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, for Reactor Vessel Examinations RR-04-GO-002 (TAC Nos. MC3804, MC3805, MC3807, and MC3810)" (ML420040261)

- NRC Safety Evaluation dated August 2, 2005, for Browns Ferry Units 1, 2 and 3; Sequoyah Nuclear Plant Units 1 and 2; and Watts Bar Nuclear Plant Unit 1, "in-service Inspection Program Relief Request PDI-4 (TAC Nos. MC6232, MC6233, MC6234, MC6235, MC6236, and MC6237)" (ML051730487)