ATTACHMENT 2

April 28, 2014

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Commission

In the Matter of	)	
	)	
Florida Power & Light Company	)	Docket No. 50-389
	)	
(St. Lucie Plant, Unit 2)	)	

DECLARATION OF MR. RUDY GIL IN SUPPORT OF FPL'S ANSWER OPPOSING SACE REQUEST FOR HEARING

Mr. Rudy Gil states as follows under penalty of perjury:

I. INTRODUCTION

A. Declarant Background

1. I am the Corporate Manager of Programs for the NextEra Energy, Inc. nuclear fleet. My educational background and qualifications include receiving a Bachelor of Science in Civil Engineering in 1978 from the University of Miami. I have held a Professional Engineer License from the State of Florida since 1982, and received a Senior Reactor Operator Management Certification in 2000. I have been employed by the Florida Power & Light Company ("FPL") Nuclear Division since 1978. I have held various engineering positions with increasing responsibility, including Corporate Chief Civil Engineer, Design Engineering Manager and Systems Engineering Manager for the St. Lucie Nuclear Plant; Assistant Maintenance Manager at the Turkey Point Nuclear Plant; and my current position as Corporate Programs Engineering Manager. In my current position, which I have

held since 1998, I am responsible for the health of the engineering programs for the NextEra Energy, Inc. nuclear fleet. This includes ownership of the steam generator program for each of our pressurized water reactor sites, including St. Lucie. I have been active in the resolution of various nuclear industry issues. Examples of significant industry committees include the Nuclear Energy Institute task forces for Reactor Head Drop Analysis, Buried Piping and Fukushima Flooding. In addition, I am a member of the Electric Power Research Institute Advisory Committee for Materials Management and the Executive Oversight Committee for the Pressurized Water Reactor Materials Management Program. These committees provide guidance and oversight for industry initiatives related to reactor coolant system materials programs, including steam generator management.

B. SACE's Request for Hearing

2. I have reviewed, and am familiar with, Southern Alliance for Clean Energy's ("SACE") hearing request regarding the alleged *de facto* amendment of St. Lucie Unit 2's operating license, which was filed with the Secretary on March 10, 2014.[1] I am also familiar with the Declaration provided by Mr. Arnold Gundersen in support of SACE's hearing request.[2]

3. My Declaration addresses SACE's claim that FPL, with the continuing acquiescence of the NRC Staff, inappropriately installed replacement steam generators in 2007 at St. Lucie Unit 2 with four changes to the safety design without a required a license amendment. Request for Hearing at 2; Gundersen Decl. at ¶ 31. I iden-

[1] Southern Alliance for Clean Energy's Hearing Request Regarding *De Facto* Amendment of St. Lucie Unit 2 Operating License (Mar. 10, 2014) ("Hearing Request").

[2] Declaration of Arnold Gundersen (Mar. 9, 2014), Attachment 1 to Southern Alliance for Clean Energy's Hearing Request Regarding *De Facto* Amendment of St. Lucie Unit 2 Operating License (Mar. 10, 2014) ("Gundersen Decl.").

tify below the description of the St. Lucie Unit 2 steam generators in the Updated Final Safety Analysis Report ("UFSAR"). I also describe how SACE mischaracterizes the four enhancements to the St. Lucie Unit 2 steam generators' design and identifies no safety issues relating to the four design changes.

4. My Declaration also addresses SACE's erroneous implication that the tube wear at St. Lucie Unit 2 is similar to tube wear that led to the shutdown of the San Onofre Nuclear Generating Station ("SONGS"). Gundersen Decl. at ¶ 36. Significant differences between the design of the St. Lucie Unit 2 and SONGS steam generators, as well as differences in the type of tube wear experienced, make such a comparison inappropriate and misleading. The fourth eddy current inspection of 100% of the in-service tubes in St. Lucie Unit 2's steam generators completed in March 2014 confirmed again that St. Lucie is not susceptible to the unique in-plane tube-to-tube wear experienced at SONGS.

II. THE FOUR ST. LUCIE UNIT 2 STEAM GENERATOR DESIGN CHANGES HIGHLIGHTED BY SACE AND MR. GUNDERSEN DO NOT RAISE SAFETY ISSUES

5. The only design changes of interest identified by SACE and Mr. Gundersen are "removal of the stay cylinder, the perforation of the central region of the tube sheet, the addition of 588 tubes in the central region, and the substitution of the broached trefoil plates for a lattice or egg crate support for the thousands of tubes in each steam generator." Hearing Request at 2; Gundersen Decl. at ¶ 31. SACE asserts that these "drastic alterations" required a license amendment "before FPL could implement them." Hearing Request at 2. SACE further claims "neither FPL nor the NRC Staff has analyzed how their removal or alternation will affect the behavior of the [reactor coolant pressure boundary] and the safety of Unit 2."

Id. Nothing could be further from the truth. Mr. William Cross's Declaration details the safety reviews and licensing actions pursuant to NRC regulations performed by FPL, the NRC Staff and the Advisory Committee on Reactor Safeguards, noting in particular the NRC License Amendment granted to operate St. Lucie Unit 2 at higher power with the replacement steam generators. I will summarize the facts as set forth in the current St. Lucie Unit 2 UFSAR regarding the steam generators and the four design changes mischaracterized by SACE and Mr. Gundersen.

A. Removal of the Stay Cylinder

6. The purpose of the stay cylinder in the Combustion Engineering design for St. Lucie Unit 2's original steam generators was to provide structural support for the large diameter tube sheet plates to meet the American Society of Mechanical Engineering ("ASME") code and regulatory requirements. AREVA's design, which was used for the replacement steam generators, instead utilizes a divider plate integrally welded to the tube sheet face and primary head to separate the inlet and outlet plenums and to provide the same structural support in accordance with ASME code and regulatory requirements. *See* Exhibit B, UFSAR at 5.4-11.

7. The divider plates are described in Section 5.4.2.1.2 of St. Lucie Unit 2's current UFSAR (Amendment 21, dated November 2012) (ADAMS Accession No. ML14084A568). However, they were first described in Amendment 18 to that UFSAR dated January 2008. Exhibit B, UFSAR at 5.4-11. The current UFSAR (and Amendment 18 to the UFSAR) also removed "stay cylinder" from the list of steam generator components in UFSAR Table 5.2-3. Exhibit C. Mr. Gundersen's

position concerning the stay cylinders relies on his Exhibits 2 and 3, which are outdated versions of the UFSAR that preceded Amendment 18.

8. The structural evaluation of the St. Lucie Unit 2 steam generators for Extended Power Uprate ("EPU") was performed by rerunning the design basis analyses relative to the 1517 MWt/SG conditions specified for power uprate. The scope of the reanalysis included the entire steam generator pressure boundary and all internal and external pressure boundary attachments. Formal analyses were performed for the tube sheet, tube bundle, primary divider plate, primary head and internal attachments, secondary shell and internal/external attachments, primary and secondary nozzles, primary and secondary manways, handholes, inspection ports, and studs and covers on all bolted openings. License Amendment Request for Extended Power Uprate, St. Lucie Unit 2, FPL L-2011-021, dated February 25, 2011, Attachment 5, EPU Licensing Report, 2.2.2-66 ("EPU LAR") (ADAMS Accession No. ML110730299). This structural analysis with the divider plate at EPU power levels was reviewed by the NRC Staff prior to the EPU license amendment. AREVA's use of the divider plate integrally welded to the tube sheet in lieu of a stay cylinder has been fully analyzed and meets all ASME code requirements.

9. Mr. Gundersen states that the stay cylinder was designed "to prevent the tube sheet from flexing upward in the event of an accident." Gundersen Decl. at ¶ 16; *see also id.* at ¶ 61. More correctly stated, the purpose of the stay cylinder was to reduce tube sheet stress levels in bending to acceptable design levels. The divider plate in the St. Lucie steam generators provides that same function. Mr. Gunder-

sen provides no information, opinion or support to the contrary. Indeed, his Exhibit 4 notes that a number of Combustion Engineering plant replacement steam generators manufactured by vendors other than AREVA also "removed" stay cylinders and instead provided divider plates similar to AREVA's design.

B. The Addition of 588 Tubes

10. Mr. Gundersen states that the addition of 588 tubes as part of steam generator replacement was a change with "significant safety implications." Gundersen Decl. at ¶ 62. He alleges that neither FPL nor the NRC performed the analyses required by such changes. Gundersen Decl. at ¶ 65.

11. The replacement steam generator design with the additional 588 tubes meets applicable codes and regulatory standards. These tubes were evaluated in AREVA's standard thermal-hydraulic, flow, and vibration analysis modelling that was used when it designed replacement steam generators for St. Lucie Unit 2 and other nuclear power plants. Mr. Gundersen provides no information, opinion, or support to the contrary.

12. The additional 588 tubes (for a total of 8,999) are described in Section 5.4.2.1.2 of St. Lucie Unit 2's current UFSAR. *See* Exhibit B, UFSAR at 5.4-11. Indeed, the existing number of tubes was first described in St. Lucie Unit 2's Amendment 18 to the UFSAR dated January 2008. The additional tubes were taken into account at the time the NRC approved the EPU license amendment request. EPU LAR at 2.2.2-58. Mr. Gundersen's position concerning the 588 additional tubes relies on his Exhibit 3, which is an outdated version of St. Lucie Unit 2's FSAR that preceded Amendment 18 and did not consider the additional tubes.

C. Holes Drilled in the Tube Sheet's Central Region to Insert Additional Tubes

13. Mr. Gundersen states that additional holes that were drilled into the tube sheet to accommodate the number of additional tubes results in a "less solid" tubesheet that has safety implications. He states that FPL and the NRC have not analyzed the public health and safety impact of this condition. Gundersen Decl. at ¶¶ 61, 65.

14. The tubesheet is not weakened by the perforations to accommodate the additional tubes. The AREVA replacement steam generators use AREVA's standard divider plate design, which analyses and testing have shown support the tubesheet such that ASME codes and regulatory requirements are met even when considering the effects on the tube sheet of additional tubes. Mr. Gundersen provides no information, opinion, or support to the contrary.

15. As I described above, the current UFSAR (as far back as Amendment 18 in January 2008) notes the new number of tubes present in the St. Lucie Unit 2 replacement steam generators and thus the additional holes drilled in the tube sheet to accommodate the tubes. The additional tubesheet holes were taken into account at the time the NRC approved the EPU license amendment. EPU LAR at 2.2.2-67. Mr. Gundersen's position again relies on an outdated version of the technical specifications.

D. The Substitution of Broached Trefoil Tube Support Plates for the Lattice or Egg Crate Support System

16. Unlike the outdated FSAR on which Mr. Gundersen relies, the current St. Lucie Unit 2 UFSAR describes the use of stainless steel broached-hole tube support plates, rather than a lattice or egg crate support system, to provide support for the

tubes and prevent denting. *Compare* Gundersen Exhibit 3 at 5.4-13 *with* Exhibit B, UFSAR at 5.4-11, 5.4-13. Like the other changes discussed above, the current UFSAR description of the tube support plates was adopted in FSAR Amendment 18, well before the NRC reviewed FPL's license amendment request for the St. Lucie Unit 2 EPU.

17. Mr. Gundersen is mistaken when he claims that, for the original steam generators, "FPL chose the egg crate design for the specific purpose of reducing the potential for tube vibration." Gundersen Decl. at ¶ 25. As noted in the FSAR for those steam generators, the egg crate tube support design was chosen to preclude denting. Gundersen Decl. Exhibit 3 at 5.4-13. All tube support plates are part of the steam generator anti-vibration support system.

18. The stainless steel material and broached holes in the replacement steam generators also preclude denting. The design of the broached holes and flat contact lands has the targeted design function of providing limited motion of the tubing. *See* EPU LAR at 2.2.2-58. The design of the tube support plates used at St. Lucie Unit 2 has been used, and continues to perform successfully, in many dozens of steam generators world-wide.

19. Mr. Gundersen has failed to provide any support for his assertion that the broached tube support plates have "increased resistance to flow" and has led to "vibrational problems." Gundersen Decl. at ¶ 26. Mr. Gundersen has not identified any safety issue in connection with the broached-hole tube support plates currently used in St. Lucie Unit 2 and in other steam generators for decades.

III. TUBE WEAR AT ST. LUCIE UNIT 2 STEAM GENERATORS AND RESULTS OF THE MARCH 2014 EDDY CURRENT INSPECTION OF 100% OF IN-SERVICE TUBES OF ST. LUCIE UNIT 2

20. This section of my Declaration provides a brief history of tube wear in the St. Lucie Unit 2 replacement steam generators and the actions undertaken to ensure safe operation of that Unit.

21. Eddy current testing inspections performed after the first cycle of operation for the St. Lucie 2 steam generators identified a number of tube-to-anti-vibration bar ("AVB") wear indications in the U-bend area. As required by FPL's Steam Generator Management program and the Unit's technical specifications, operational assessments for the second cycle of plant operation were prepared. Based on those assessments, no tubes required plugging for the second cycle. However, FPL conservatively plugged all tubes (14) with a wear level greater than 25%.

22. Inspections performed after the second cycle of operation identified an increased number of tube wear indications, as predicted by the operational assessments. Wear rates, however, decreased during the second cycle, also as predicted. Based on the operational assessment prepared after the second cycle, one tube required plugging. FPL conservatively plugged an additional 20 tubes.

23. As predicted by the operational assessment performed after the second cycle, inspections performed after the third cycle again identified an increased number of indications. Wear rates, however, continued to decrease, also as predicted. The indications were limited to the same three regions of the tube bundle where indications were found after the first and second cycles. Seven tubes required plugging during this inspection, and FPL conservatively plugged an additional 112 tubes.

24. These inspections found no tube-to-tube wear in St. Lucie Unit 2's steam generators. Only tube-to-AVB wear occurred.

25. An inspection of 100% of the in-service tubes in St. Lucie Unit 2 steam generators after the first cycle of operation with power uprate (end of fourth cycle) was completed in March 2014. The inspection data continues to be reviewed and verified. The following preliminary conclusions have been reported to the NRC: (a) steam generator wear continues to be manageable within the St. Lucie Unit 2 steam generator program; (b) no tube integrity issues were identified; (c) the number of tubes plugged (69) was significantly less than the number of tubes plugged during the last outage prior to power uprate; most plugging was implemented to increase margin rather than because it was required by technical specifications; (d) the tube wear was not unexpected; average and statistical wear rates were at approximately 2012 levels notwithstanding power uprate; (e) again, no in-plane tube-to-tube wear was detected; and (f) no issues were identified based on the SONGS operating experience. NRC inspectors were on-site during the inspection of tubes and the results were communicated to NRC Region II. The preliminary operational assessment supports full cycle operation, after which the steam generators will be inspected again.

26. The root cause of the tube wear at St. Lucie Unit 2 is well understood. The comprehensive root cause evaluation conducted by AREVA for FPL, and reviewed by an independent third party, concluded that the U-tubes were not effectively supported during the tubing installation process. The tube bundle was allowed to sag, causing slight deformation of the AVBs, which, in turn, closed the tube-to-AVB

gap. This condition caused tubes to be in constant contact with AVBs at several locations, contrary to the design specification and models. The resulting tube-to-AVB contact allowed tube wear in the affected tubes/areas. However, as expected based on this root cause, the tube wear has attenuated over time and is manageable under FPL's Steam Generator Management program.

IV. THERE ARE SIGNIFICANT DESIGN DIFFERENCES BETWEEN ST. LUCIE AND SONGS

27. SACE is wrong when it implies that the situation leading to the shutdown of SONGS Units 2 and 3 is similar to events at St. Lucie Unit 2. SACE Motion at 2; Gundersen Decl. at ¶ 36.

28. Tube-to-tube wear in the in-plane direction caused the SONGS tube leak that culminated in that plant's shutdown. In contrast, after four full cycles of operation, St. Lucie Unit 2 has not experienced tube-to-tube wear, and there is no reason to believe that such wear will occur at St. Lucie Unit 2.

29. The type of tube-to-tube wear that occurred at SONGS had never been seen before in a commercial nuclear power plant. In contrast, the tube-to-AVB wear experienced at St. Lucie Unit 2 is well-known and understood in the nuclear industry. As a result, FPL was able to rely on well-developed industry standards that have been proven successful, over many, many years of cumulative industry experience, to address tube-to-AVB wear and to ensure safe operation of St. Lucie Unit 2.

30. The cause of the tube leak that occurred at SONGS is not a concern at St. Lucie Unit 2, in large part because of various design differences between the steam generators at those plants, which were designed and built by different vendors. Ex-

hibit A to this Declaration provides a detailed list of those differences. Due to these differences, the St. Lucie Unit 2 steam generators have not experienced, and are not expected to experience, the high steam velocity and void fraction (steam to water ratio) conditions that led to the in-plane "fluid elastic" vibration of steam generator tubes at SONGS.

I declare under penalty of perjury that the foregoing is true and correct.

<u>Executed in Accord with 10 C.F.R. § 2.304(d)</u>
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Exhibit A

The differences between the St. Lucie Unit 2 and SONGS steam generators include:

1. The steam generators are designed and built by different vendors;
2. The retainer bar feature that resulted in tube wear for the SONGS steam generators is not included in the St. Lucie Unit 2 design;
3. The AVB systems are different (number of sets and arrangement), which leads to different flow patterns;
4. Maximum void fraction in the SONGS steam generators is higher than that of St. Lucie Unit 2;
5. Based on nominal plant power output, it can be concluded that the mean kinetic energy at the U-bend outlet for St. Lucie Unit 2 is significantly less than at SONGS, even after implementation of the power uprate;
6. The St. Lucie Unit 2 tube lane width is minimized, which positively impacts the by-pass flow, which can exist in this particular area without tubes;
7. The St Lucie Unit 2 tube lane area of each Tube Support Plate presents not only rectangular shape holes but also drilled holes that can favor the mixing between fluid in the tube lane and fluid in the tube bundle;
8. Considering items 3, 4, and 5 above, the SONGS flow pattern departs from the classical flow pattern expected in the U-bend of an AREVA steam generator.

Subsection 5.2.3.3.1. Fracture toughness data for the steam generator materials is presented in Tables 5.2-8, 5.2-10, and 5.2-12.

During final assembly and shipment, the steam generator primary and secondary sides are brought to a state of cleanliness consistent with the rest of the fluid system in interfaces with as described in Subsection 5.2.3. In addition, the interior of the steam generator is protected by pressurizing with an inert gas during shipment and interim storage. Cleanliness during construction is discussed in Subsection 5.2.3.4.1.2.1.

The chemistry control and corrosion control effectiveness of the secondary side water is discussed in Subsection 10.3.5.

5.4.2.1.2 Steam Generator Description

The nuclear steam supply system utilizes two steam generators (Figure 5.4-6) to transfer the heat generated in the Reactor Coolant System to the secondary system. The design parameters for the steam generators are given in Table 5.4-2.

The steam generator is a vertical U-tube heat exchanger with the reactor coolant on the tube side and the secondary fluid on the shell side.

Reactor coolant enters the steam generator through the 42 inch ID inlet nozzle, flows through 3/4 inch OD 0.0429 inch wall U-tubes, and leaves through two 30 inch ID outlet nozzles. Divider plates in the lower head separate the inlet and outlet plenums. The plenums are carbon steel with stainless steel clad. The reactor coolant side of the tube sheet is Ni-Cr-Fe clad. The U-tubes are Inconel 690 composition.

The steam generator contains 8999 U-tubes for heat transfer for primary to secondary water. Each tube is expanded into the tube sheet so that there is no voids or crevices occurring along the entire length of the tube sheet interface. The tubes are also welded to the Ni-Cr-Fe alloy clad on the reactor coolant surface of the tubesheet. The tube to tubesheet welding conforms with the requirements of the ASME Code, Sections III and IX. Support for the tube bundles are by stainless steel broached tube support plates. Additional support is provided by stainless steel anti-vibration bars to prevent excessive flow-induced vibration.

Feedwater enters the steam generator through the feedwater nozzle where it distributed via a feedwater distribution ring. The feedwater ring is constructed with discharge nozzles which are configured in the form of a "J". These nozzles are welded to the top of the ring and direct the feedwater flow away from the shell. This construction greatly reduces the rate at which the ring drains, helping to provide assurance that the feedwater ring remains full of water as long as there is feedwater flow when the level in the steam generator drops below the feedwater ring (refer to Figures 5.4-6, 16, and 17).

The downcomer in the steam generator is an annular passage formed by the inner surface of the steam generator shell and the cylindrical shell that encloses the vertical U-tubes. Upon exiting from the bottom of the downcomer, the secondary flow is directed upward over the vertical U-tubes. Heat transferred from the primary side converts a portion of the secondary flow into steam.

Upon leaving the vertical U-tube heat transfer surface, the steam-water mixture enters the cyclone-type separators. These impart a centrifugal motion in the mixture and separate the water particles from the steam. The water exists from the perforated separator housing and combines with the feedwater to repeat the cycle. Final drying of the steam is accomplished by passage of the steam through the double pocket, chevron-type dryers.

The steam generators are mounted on bearing plates which allow controlled lateral motion due to thermal expansion of the reactor coolant piping. Key stops embedded in the concrete base limit this motion in case of a reactor coolant pipe rupture. The top of each unit is restrained from sudden lateral movement by keys and hydraulic snubbers mounted rigidly in the concrete structure.

The steam generators are located at a higher elevation than the reactor vessel. The elevation difference created natural circulation capability sufficient to remove core decay heat following coast down of all reactor coolant pumps.

Overpressure protection for the shell side of the steam generators and the main steam line up to the inlet of the turbine stop valve is provided by 16 flanged spring loaded ASME Code safety valves which discharge to atmosphere. Overpressure protection is discussed in Subsection 5.2.2.

5.4.2.1.3 Steam Generator Tubes

The steam generator are tubed with 0.750 inch OD by .0429 wall tubes. The tubes are fabricated from Inconel 690 to insure compatibility with both the primary and secondary waters. The design incorporates a general corrosion allowance that provided for reliable operation over the plant design lifetime.

Localized corrosion has led to steam generator tube leakage in some operating plants. Examination of tube defects that have resulted in leakage has shown that two mechanisms are primarily responsible. These localized corrosion mechanisms are referred to as (1) stress assisted caustic cracking, and (2) wastage or beavering. Both of these types of corrosion have been related to steam generators that have operated on phosphate chemistry. The caustic stress corrosion type of failure is precluded by controlling feedwater chemistry to the specification limits shown in Subsection 10.3.5. Removal of solids from the secondary side of the steam generator is discussed in Subsection 10.4.8. Localized wastage or beavering has been eliminated by removing phosphates from the chemistry control system.

Volatile chemistry (discussed in Subsection 10.3.5) has been successfully used in all CE steam generators that have gone into operation since 1972.

a) Tube Degradation Mechanisms

The design steady state and transient conditions specified in the design of the steam generator tubes are discussed in Subsection 3.9.1.1.

Alloy 690 tubes are less susceptible to various forms of the degradation.

NRC Information Notice 90-49, "Stress Corrosion Cracking of PWR Steam Generator Tubes," identified conditions of similar steam generators, where circumferential cracks were observed near the tube expansion transition at or near the top of the tubesheet.

Should a circumferential crack be detected the tube may have a stabilizer and tube plug installed. If the plugged tube severs, the stabilizer is designed to reduce the possibility of tube-to-tube contact. The stabilizers the plugs and their installation are designed to function under all operating, transient or test conditions of the steam generator. This installation takes into consideration maintaining integrity under vibrating loads and material compatibility with tube material subject to both reactor coolant and feedwater system environments.

A number of operating plants have experienced a corrosion phenomenon known as "denting".

Denting is caused by the uncontrolled corrosion of carbon steel support structure surfaces surrounding a tube. As the uncontrolled corrosion of carbon steel takes place, the original base metal (iron) is converted to nonprotective magnetite (Fe_3O_4) resulting in a doubling of volume (i.e., twice the volume of the original base metal is occupied by the metal oxide). Because the magnetite is nonprotective, the base metal continues to corrode, producing large localized concentrations of metal oxide. The expanded metal oxide exerts pressure on the steam generator tube and the support. Then pressure in the tube/tube support annulus becomes sufficient to produce yielding in the tube wall, denting results.

Experience from operating steam generators and laboratory testing has demonstrated that two conditions are required to initiate denting:

1) The original clearance between the tube and the support must have become blocked with a porous deposit in which bulk water can enter and be concentrated.

2) The bulk water being concentrated must have impurities that produce chloric acid solutions, which in corroding the carbon steel of the support result in the formation of a nonprotective form of magnetite.

The potential for tube denting has been reduced in the St. Lucie Unit 2 steam generators by the installation of tube support plates and antivibration bars system that are stainless steel with a high chromium content that forms a tight adherent oxide layer. This combination eliminates the potential for denting.

Exhibit C

TABLE 5.2-3

REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

Component	Material Specification
Reactor vessel	
Shell	SA-533 Grade B, Class 1 Steel
Forgings	SA-508 Class 1 and 2
Cladding[(a)]	Weld deposited austenitic stainless steel with greater than 5% delta ferrite (Equivalent to SA-240 Type 304) or NiCrFe alloy (equivalent to SB-168)
Replacement Reactor Vessel Closure Head (RVCH)	
Forging	SA-508, Class 3
Replacement RVCH Cladding	Weld Deposited austenitic stainless steel:
	First layer is 309L with delta ferrite number acceptable range of 5FN to 20FN. Subsequent layers are 308L with delta ferrite number acceptable range of 5FN to 17.6FN.
CEDM Nozzles	Nozzle: SB-167 (Alloy 690)
	Adapter: SB-166 (Alloy 690)
	Weld Filler: 52 or 52M (Alloy 690)
Instrument Nozzle	Nozzle: SB-167 (Alloy 690)
	Adapter: SA-479 Type 304 Stainless Steel
	Weld Filler: 52 or 52M (Alloy 690)
Vessel Internals[(a)]	Austenitic Stainless Steel and NiCrFe alloy
Fuel Cladding[(a)]	Zircaloy-4
Control element drive mechanism housings	
Lower	SA-182 Type 403 stainless steel Special Code Case 1334 with end fittings to SB-166 (Alloy 690)
Upper	SA-479 and SA-213 Type 316 stainless steel with end fitting of SB-166 and vent valve seal of Type 440 stainless steel seat to SA-479
Closure head bolts & Nuts	SA-540 B23 and B24, Class 3
Support (on Nozzles)	SA-508, Class 2
Pressurizer -	
Shell	SA-533 Grade B Class 1
Upper Head Instrument Nozzle	SA-533 Grade B Class 1
Penetration Bore[(a)]	
Cladding[(a)]	Weld deposited austenitic stainless steel with greater than 5% delta ferrite or NiCrFe alloy (equivalent to SB-168)

(a) Materials exposed to reactor coolant

(b) Special weld wire with low residual elements of copper and phosphorus is specified for the reactor vessel core beltline region.

(c) The four (4) one-inch instrument nozzles in the upper head have SA-182, Type 316L safe ends.

TABLE 5.2-3 (Cont'd)

Component	Material Specification
Pressurizer (Cont'd)	
Forged nozzles	SA-508 Class 2
Instrument nozzles[(a)]	SB-166
Surge and PORV	
nozzle safe ends[(a)]	SA-351, Gr CF8M
Spray and instrument	SA-182, Type 316, except upper head
nozzle safe ends[(a)(c)]	instrument nozzles have Type F316L
Studs and nuts	SA-540 Grade B24 and SA-194 Grade 7
Steam generator	
Primary head, nozzles and manways	SA-508 Grade 3, Class 2
Primary divider plate	SB-168 UNSN 06690
Primary nozzle safe ends	SA-105
Primary head cladding[(a)]	Weld deposited austenitic stainless steel (Type 308L and 309L)
Tubesheet	SA-508 Grade 3, Class 2
Tubesheet cladding	Weld deposited NiCrFe (Alloys 52 / 152)
Tube[(a)]	NiCrFe Alloy (SB-163 UNSN 06690)
Secondary shell and head	SA-508 Grade 3, Class 2
Secondary nozzles	SA-508 Grade 3, Class 2
Steam nozzle venturis	SB-166 UNSN 06690
Secondary instrument nozzles	SA-105
Studs / Nuts	SA-193 B16 / SA-194 Grade 7
Support Skirt	SA-508 Grade 3, Class 2
Sliding base support studs and nuts	SA-540 Grade B23 Class 2

TABLE 5.2-3 (Cont'd)

Component	Material Specification
Reactor coolant pumps	
Casing[(a)]	SA-531 Grade CF8M
Internals[(a)]	Austenitic stainless steel (SA-351 Grade CF81 ASTM-A-479 Type 316 ASTM-A-240 Type 316, SA-182 GR F 304)
Studs and Nuts	SA-540 Grade D23 and SA-194 Grade B-7
Reactor coolant piping	
Pipe (30 in. and 42 in.)	SA-516 Grade 70
Cladding[(a)]	SA-240 - 304L
Surge Line (12 in.)[(a)]	SA-351 - CF8M
Spray Line Pipe	SA-312, Type 316
	SA-312, 304L
Spray Line Fittings	SA-403, Type WP 316 SA-182, F 316 SA-376, TP 316 SA-182(M), TP 316/316L SA-182, F304L SA-182, F316/F316L
Piping safe ends (30 in.)[(a)]	SA-351 - CF8M
Surge nozzle forging	SA-541-1
Surge nozzle safe end[(a)]	SA-351 - CF8M
Shutdown cooling outlet nozzle forgings	SA-541-1
Shutdown cooling outlet[(a)] nozzle safe ends	SA-351 - CF8M
Safety injection nozzle forgings	SA-182 - F1
Safety injection nozzle[(a)] safe ends	SA-351 - CF8M
Charging inlet nozzle forging	SA-182 - F1
Charging inlet nozzle[(a)] safe end	SA-182 - F316
Spray nozzle forgings	SA-105 Grade II
Letdown and drain pipe	SA-312, 304L

TABLE 5.2-3 (Cont'd)

Component	Material Specification		
Reactor coolant piping (Cont'd)			
Spray nozzle safe ends [a]	SA-479, 316L or SA-182, F316L		
Letdown and drain or drain nozzle forgings	SA-105 - Grade II		
Letdown and drain or drain nozzle safe ends[a]	SA-479, 316L or SA-182, F316L		
Sampling or pressure[a] measurement nozzles	SB-166		
Sampling or pressure measurement nozzle safe ends[a]	SA-182 - F316 or SA-479 TP-316		
RTD nozzles[a]	SB-166		
Hot Leg RTD Split Nozzle Penetration Bore[a]	SA-516 Grade 70		
Sampling nozzle (surge line)[a]	SA-182 - F316		
RTD nozzle (surge line)[a]	SA-182 - F316		
Nozzle Thermal Sleeves[a]	SB-166 or SB168		
Valves[a]	SA-351 - CF8M, SA-182-F316		
AE - Supplied Components			
Valves	ASME	SA-182 SA-479	(F-316), SA-564 (Type-630) (Type-347, 348, 316L)
Pipes	ASME	SA-312	(GR TP 304)
Fittings	ASME	SA-182 SA-403 SA-351	(F-304) (GR-WP 304W) (GR CF8)
Flanges	ASME	SA-182 SA-351	(F-304) (GR-CF8)
Restrictors	ASME	SA-182	F-316
Bolts, Nuts	ASME	SA-193 SA-194	GR B7 GR 2H