UNITED STATES OF AMERICA
BEFORE THE NUCLEAR REGULATORY COMMISSION

In the Matter of:	)	
Florida Power & Light Co.	)	Docket No. 50-389
St. Lucie Plant, Unit 2	)	April 25, 2014

SOUTHERN ALLIANCE FOR CLEAN ENERGY'S AMENDED HEARING REQUEST REGARDING *DE FACTO* AMENDMENT OF ST. LUCIE UNIT 2 OPERATING LICENSE

I. INTRODUCTION

On March 10, 2014, the Southern Alliance for Clean Energy ("SACE") submitted a request for a hearing and two contentions regarding the U.S. Nuclear Regulatory Commission ("NRC") Staff's *de facto* amendment of the St. Lucie Unit 2 operating license to allow operation of the reactor with substantially re-designed steam generators. Hearing Request Regarding De Facto Amendment of St. Lucie Unit 2 Operating License ("Hearing Request"). The Hearing Request was supported by the Declaration of Arnold Gunderson (March 9, 2014) ("Gundersen Declaration"). Pursuant to 10 C.F.R. §§ 2.309(c)(1), SACE hereby amends its Hearing Request to provide additional relevant information that is contained in Amendment 18 to the Updated Final Safety Analysis Report ("UFSAR Amendment 18") (June 26, 2008) (ML14104B631). The UFSAR Amendment 18 was not publicly available at the time that SACE submitted its Hearing Request but was released by the NRC's Public Document Room ("PDR") on April 15, 2014. The newly disclosed information supports SACE's assertion in Contentions 1 and 2 that the changes to the St. Lucie steam generators that were made by Florida Power & Light Co. ("FPL") in 2007 constitute major design changes that exceed the reactor's design basis as described in the original 1980 Final Safety Analysis Report ("OFSAR").

This Amended Hearing Request is supported by the attached Supplemental Declaration of Arnold Gundersen (April 24, 2014) ("Supplemental Gundersen Declaration") (attached as Exhibit 1).

II. NEW INFORMATION SUPPORTING CONTENTIONS 1 AND 2

SACE hereby amends its Hearing Request to provide the following additional information from UFSAR Amendment 18:

1. As stated in SACE's Hearing Request, in 2007, FPL replaced the Unit 2 original steam generators ("OSGs") with new Replacement Steam Generators ("RSGs") manufactured by Areva. *Id.* at 7. In June of 2008, pursuant to 10 C.F.R. § 50.59, FPL filed a report with the NRC that summarized the characteristics of the RSGs and asserted that FPL had made "no significant changes to major component supports or piping supports." *Id.* at 7-8 (citing St. Lucie Unit 2, Docket No. 50-389, Changes, Tests, and Experiments Made as Allowed by 10 C.F.R. 50.59 for the Period of June 12, 2006 through April 4, 2008 at 8 (attached to letter from Gordon L. Johnston, FPL, to NRC re: St. Lucie Unit 2 Docket No. 50-389 Report of 10 CFR 50.59 Plant Changes (June 26, 2008)) (ML081840111) ("50.59 Summary")). At the same time that FPL submitted the 50.59 Summary to the NRC, it also submitted UFSAR Amendment 18, which updates the FSAR to reflect the changes made by FPL under 10 C.F.R. § 50.59.
2. In UFSAR Amendment 18, changes to the FSAR that resulted from Amendment 18 are marked by vertical lines in the right-hand margin of the pages. These changes include only additions or substitutions of text and do not include strikeouts. Therefore, in order to understand the nature of the changes, it is necessary to compare UFSAR Amendment 18 with

the OFSAR that FPL submitted in 1980 (relevant pages of Chapter 5 of the OFSAR are attached as Exhibit 2.)

3. The change pages in UFSAR Amendment 18 confirm that FPL removed or altered components of the steam generators that are significant to the safe operation of St. Lucie Unit 2 in the following respects:

a) In contrast to the OFSAR, UFSAR Amendment 18 no longer identifies the stay cylinder (*i.e.,* "Tubesheet stay") as a component of the Unit 2 RSGs. *Compare* Table 5.2-3 in OFSAR (page 5.2-27) with Table 5.2-3 in UFSAR Amendment 18 (page 5.2-29). It is reasonable to infer from this omission that the stay cylinder has been removed from both RSGs. *See* Gundersen Declaration, ¶ 31.

b) UFSAR Amendment 18 confirms that the RSGs have 588 new steam generator tubes in addition to the 8,411 tubes in the OSGs, totaling 8,999 tubes. *Compare* OFSAR § 5.4.2.1.2 at 5.4-11 with UFSAR Amendment 18 § 5.4.2.1.2 at 5.4-11. The addition of 588 new tubes changes the pattern of water circulation in the steam generator and therefore has significant safety implications. *See* Gundersen Declaration, ¶ 62.

c) In addition, it is reasonable to infer that in order to accommodate the 588 additional tubes, the tubesheet in the RSGs contains 588 additional perforations. These additional perforations increase the potential for tubesheet flexing. *See* Gunderson Declaration, ¶¶ 31 and 61.

d) UFSAR Amendment 18 deletes all references to the eggcrate tube supports that FPL relied on in the OFSAR for the purpose of avoiding denting of tubes. Instead, Amendment 18 now states that FPL uses plate supports in order to avoid tube denting. *Compare* the text of the 1980 OFSAR § 5.4.2.1.3 with UFSAR Amendment 18 §

5.4.2.1.3 as follows:

> In the OFSAR, FPL stated:
>
> The potential for tube denting has been reduced in the St. Lucie Unit 2 steam generators by the installation of an antivibration support system that does not use drilled support plates. Supports of the same type, "egg crates", have been used to some extent in all of supplied commercial steam generators within the United States.
>
> The egg crate system reduces susceptibility to tube denting by providing larger clearances and increased flow area around the tubes, so that the clearances between the tubes and their supports are less likely to become plugged by corrosion products.
>
> St. Lucie Unit 2 has a full egg crate support system (all support plates have been eliminated).

Id. at 5.4-13. Thus, FPL clearly stated that it would eliminate the use of support plates in order to avoid tube denting.

In 2008, in FSAR Amendment 18, FPL replaced the above analysis with the following analysis:

> 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.

Id. at 5.4-13. In other words, FPL now purports to avoid tube denting with the very same components it previously disavowed as *contributors* to tube denting. But FPL provides no explanation for this complete turnaround in its safety analysis. In fact, the substitution of broached plates for egg crate tube supports creates potential for greater vibration of tubes. Gundersen Declaration, ¶¶ 44, 45, and 61.

4. UFSAR Amendment 18 identifies a new component not previously identified in the OFSAR or described in the 50.59 Summary: "steam nozzle venturis." *Id.,* Table 5.2-3 at page 5.2-29. As discussed in the attached Supplemental Gundersen Declaration, the purpose of steam nozzle venturis is to limit the rate at which steam (mass and energy) leaves a steam generator. The existence of a different mass and energy flow rate in the RSGs would also require FPL to perform a new and different accident analysis for the steam generators. Thus, the installation of steam nozzle venturis indicates that an important safety parameter has been changed between the OSG and RSG, resulting in reanalysis and modification from the original design. The installation of this additional component should have been identified in the 50.59 analysis and should have resulted in a license amendment. *Id.,* ¶ 7.

III. CONCLUSION

In conclusion, the information described in Section II above confirms that (a) FPL has made major design changes to the St. Lucie Unit 2 steam generators that increase the risk of steam generator failure at the reactor. Therefore the information supports Contentions 1 and 2.

Respectfully submitted,

(Electronically signed by)
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April 25, 2014

EXHIBIT 1

UNITED STATES OF AMERICA
BEFORE THE NUCLEAR REGULATORY COMMISSION

In the Matter of:	)	
Florida Power & Light Co.	)	Docket No. 50-389
St. Lucie Plant, Unit 2	)	April 25, 2014

SUPPLEMENTAL DECLARATION OF ARNOLD GUNDERSEN

Under penalty of perjury, I, Arnold Gundersen, hereby declare as follows:

I. INTRODUCTION

1. My name is Arnold Gundersen. I am Chief Engineer for Fairewinds Associates, a paralegal services and expert witness firm. I have been retained by Southern Alliance for Clean Energy (SACE) to evaluate safety and licensing issues related to the replacement steam generators (RSGs) that Florida Power & Light Co. (FPL) installed in the Unit 2 St. Lucie nuclear reactor in 2007.

2. On March 9, 2014, I prepared a declaration in support of SACE's Hearing Request Regarding De Facto Amendment of St. Lucie Unit 2 Operating License (March 10, 2014), including two contentions challenging the lawfulness of the U.S. Nuclear Regulatory Commission's *de facto* amendment of the operating license for St. Lucie Unit 2 to allow operation of Unit 2 with substantially re-designed steam generators. The statements of fact I made in that declaration continue to be true to the best of my knowledge, and the statements of my professional opinion in that declaration continue to be accurate expressions of my best professional judgment.

3. The purpose of this Supplemental Declaration is to support SACE's Amended Hearing Request (April 24, 2014).

4. As summarized in ¶ 31 of my March 9 Declaration, my review of correspondence and documents related to St. Lucie Unit 2 and the San Onofre steam generators shows that FPL has made at least four major design changes to the steam generators for St. Lucie Unit 2.

First, the RSGs no longer contained the stay cylinders that were part of the original steam generator ("OSG") design discussed in the Final Safety Analysis Report ("FSAR") as structural support for the reactor coolant system and included in the Aging Management Program (AMP). Second, documents related to subsequent inspections of the St. Lucie Unit 2 steam generators show that AREVA added 588 new tubes to the original 8,411 tubes, now totaling 8,999 tubes. The addition of 588 new tubes changes the pattern of water circulation in the steam generator and therefore has significant safety implications. Third, FPL replaced the pre-existing eggcrate tube supports with trefoil broached plates," despite the fact that such plates were specifically excluded from the original steam generator design for safety reasons. Finally, in order to accommodate the 588 new tubes, it is reasonable to infer that the region of the tubesheet that had been directly above the stay cylinder was now perforated with 588 new holes.

5. As discussed in my March 9 Declaration, all of these changes have major safety significance and exceed the reactor's design basis. In addition, these design changes increase the risk of steam generator failure and therefore have an adverse effect on public health and safety.

6. At the time I prepared my March 9 Declaration, I did not have access to Amendment 18 of the updated Final Safety Analysis Report that FPL submitted to the NRC after it installed the RSGs ("UFSAR Amendment 18") (June 26, 2008) (ML14104B631). On April 15, 2014, the NRC's Public Document Room made that document publicly available. By comparing the text of the original 1980 FSAR ("OFSAR") with the text of UFSAR Amendment 18, I was able to confirm that FPL did indeed make all of the design changes described in ¶ 4 above. These changes to the text of the FSAR are discussed in more detail in SACE's Amended Hearing Request.

7. In addition, as discussed in par. 4 of SACE's Amended Hearing Request, UFSAR Amendment 18 shows that FPL made an additional design change I was not previously aware of: it installed "steam nozzle venturis." The purpose of a steam nozzle venturi is to limit the rate at which steam (mass and energy) leaves the RSG in the event of a steam line break accident. The fact that FPL included this new component in the RSG design demonstrates that the mass and energy flow rate from the RSGs is greater than the mass and energy flow rate from the OSGs. The existence of a different mass and energy flow rate in the RSGs would also require FPL to perform a new and different accident analysis for the steam generators. Thus, the installation of steam nozzle venturis indicates that an important safety parameter has been changed between the OSGs and RSGs, resulting in reanalysis and modification from the original design. The installation of this additional component should have been identified in the 50.59 analysis and should have resulted in a license amendment.

8. In conclusion, UFSAR Amendment 18 confirms that FPL has made major design changes to the steam generators for St. Lucie Unit 2 which exceed the reactor's design basis and increase the risk of steam generator failure at the reactor.

Under penalty of perjury, I declare that the foregoing statements of fact are true and correct to the best of my knowledge and that the foregoing statements of my opinion are based on my best professional judgment.

(Electronically signed pursuant to 10 C.F.R. § 2.304(d)(1))

Arnold Gundersen, MENE, RO
Fairewinds Associates, Inc
Burlington, Vermont 05401

Date: April 25, 2014

EXHIBIT 2

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REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

CHAPTER 5

TABLE OF CONTENTS

Section	Title	Page
5.0	REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS	5.1-1
5.1	SUMMARY DESCRIPTION	5.1-1
5.1.1	SCHEMATIC FLOW DIAGRAM	5.1-2
5.1.2	PIPING AND INSTRUMENT DIAGRAM	5.1-2
5.1.3	ELEVATION DRAWINGS	5.1-2
5.2	INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY (RCPB)	5.2-1
5.2.1	COMPLIANCE WITH CODES AND CODE CASES	5.2-1
5.2.2	OVERPRESSURIZATION PROTECTION	5.2-2
5.2.3	REACTOR COOLANT PRESSURE BOUNDARY MATERIAL	5.2-4
5.2.4	INSERVICE INSPECTION AND TESTING OF REACTOR COOLANT PRESSURE BOUNDARY	5.2-13
5.2.5	DETECTION OF LEAKAGE THROUGH REACTOR COOLANT PRESSURE BOUNDARY	5.2-16
5.2A	OVERPRESSURE PROTECTION FOR ST LUCIE UNIT 2 - PRESSURIZED WATER REACTOR	5.2A-1
5.3	REACTOR VESSEL	5.3-1
5.3.1	REACTOR VESSEL MATERIALS	5.3-1
5.3.2	PRESSURE-TEMPERATURE LIMITS	5.3-12
5.3.3	REACTOR VESSEL INTEGRITY	5.3-15
5.3	REFERENCES	5.3-18
5.4	COMPONENT AND SUBSYSTEM DESIGN	5.4-1
5.4.1	REACTOR COOLANT PUMPS	5.4-1
5.4.2	SYSTEM GENERATORS	5.4-10
5.4.3	REACTOR COOLANT PIPING	5.4-15

CHAPTER 5

TABLE OF CONTENTS (Cont'd)

Section	Title	Page
5.4.4	MAIN STEAM LINE FLOW RESTRICTIONS	5.4-18
5.4.5	MAIN STEAM ISOLATION SYSTEM	5.4-18
5.4.6	REACTOR CORE ISOLATION COOLING SYSTEM	5.4-18
5.4.7	RESIDUAL HEAT REMOVAL SYSTEM	5.4-19
5.4.8	REACTOR WATER CLEANUP SYSTEM	5.4-28
5.4.9	MAIN STEAM LINE AND FEEDWATER PIPING	5.4-29
5.4.10	PRESSURIZER	5.4-29
5.4.11	QUENCHTANK (PRESSURIZER RELIEF DISCHARGE SYSTEM)	5.4-32
5.4.12	VALVES	5.4-34
5.4.13	PRESSURIZER SAFETY AND RELIEF VALVES	5.4-35
5.4.14	COMPONENT SUPPORTS	5.4-36

REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

CHAPTER 5

LIST OF TABLES

Table	Title	Page
5.1-1	DESIGN PARAMETERS OF REACTOR COOLANT SYSTEM	5.1-3
5.1-2	REACTOR COOLANT SYSTEM VOLUMES	5.1-4
5.1-3	PROCESS DATA POINT TABULATION	5.1-5
5.2-1	REACTOR COOLANT SYSTEM PRESSURE BOUNDARY CODE REQUIREMENTS	5.2-24
5.2-2	CODE CASE INTERPRETATIONS	5.2-26
5.2-3	REACTOR COOLANT PRESSURE BOUNDARY MATERIALS	5.2-28
5.2-4	REACTOR COOLANT PRESSURE BOUNDARY WELDING MATERIAL	5.2-32
5.2-5	CHEMICAL ANALYSES OF PLATE MATERIAL IN ST LUCIE UNIT 2 REACTOR VESSEL BELTLINE	5.2-34
5.2-6	CHEMICAL ANALYSES OF WELD MATERIAL IN ST LUCIE 2 REACTOR VESSEL BELTLINE	5.2-35
5.2-7	REACTOR VESSEL TOUGHNESS PROPERTIES	5.2-37
5.2-8	STEAM GENERATOR (PRIMARY SIDE) TOUGHNESS PROPERTIES (UNIT NO. 2A)	5.2-38
5.2-9	PRESSURIZER TOUGHNESS PROPERTIES	5.2-39
5.2-10	STEAM GENERATOR (SECONDARY SIDE) TOUGHNESS PROPERTIES (UNIT 1)	5.2-40
5.2-11	STEAM GENERATOR (SECONDARY SIDE) TOUGHNESS PROPERTIES (UNIT 2)	5.2-41
5.2-12	PIPING TOUGHNESS PROPERTIES	5.2-42
5.2-13	BOLTING MATERIALS TOUGHNESS PROPERTIES	5.2-44
5.2-14	REACTOR COOLANT LEAK DETECTION SENSITIVITY	5.2-45
5.3-1	REACTOR VESSEL BELTLINE PLATE TOUGHNESS PROPERTY PREDICTIONS	5.3-19

CHAPTER 5

LIST OF TABLES (Cont'd)

Table	Title	Page
5.3-2	REACTOR VESSEL BELTLINE WELD TOUGHNESS PROPERTIES	5.3-20
5.3-3	TOTAL QUANTITY OF SPECIMENS	5.3-21
5.3-4	TYPE AND QUANTITY OF SPECIMENS FOR UNIRRADIATED TESTS	5.3-22
5.3-5	TYPE AND QUANTITY OF SPECIMENS FOR IRRADIATION EXPOSURE AND IRRADIATED TESTS	5.3-23
5.3-6	MATERIALS FOR NEUTRON THRESHOLD DETECTORS	5.3-24
5.3-7	COMPOSITION AND MELTING POINTS OF MATERIALS FOR TEMPERATURE MONITORS	5.3-25
5.3-8	TYPE AND QUANTITY OF SPECIMENS CONTAINED IN EACH IRRADIATION CAPSULE ASSEMBLY	5.3-26
5.3-9	CAPSULE ASSEMBLY REMOVAL SCHEDULE	5.3-27
5.3-10	REACTOR VESSEL DESIGN PARAMETERS	5.3-28
5.4-1	REACTOR COOLANT PUMP PARAMETERS	5.4-37
5.4-2	STEAM GENERATOR PARAMETERS	5.4-39
5.4-3	REACTOR COOLANT PIPING PARAMETERS	5.4-40
5.4-4	SHUTDOWN COOLING HEAT EXCHANGER DATA	5.4-41
5.4-5	FAILURE MODES AND EFFECTS ANALYSIS - SHUTDOWN COOLING SYSTEM	5.4-42
5.4-6	PRESSURIZER PARAMETERS	5.4-44
5.4-7	QUENCH TANK PARAMETERS	5.4-45
5.4-8	PRESSURIZER SAFETY VALVE PARAMETERS	5.4-46
5.4-9	POWER OPERATED RELIEF VALVE PARAMETERS	5.4-47

REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

CHAPTER 5

LIST OF FIGURES

Figure	Title
5.1-1	REACTOR COOLANT SYSTEM ARRANGEMENT - PLAN
5.1-2	REACTOR COOLANT SYSTEM ARRANGEMENT - ELEVATION
5.1-3	REACTOR COOLANT SYSTEM P&I DIAGRAM - SHEET 1
5.1-4	REACTOR COOLANT SYSTEM P&I DIAGRAM - SHEET 2
5.1-5	REACTOR COOLANT PRESSURE BOUNDARY DIAGRAM
5.1-6	REACTOR COOLANT PUMP P&I DIAGRAM
5.2-1	PRESSURIZER SAFETY VALVE PIPING
5.2-2	CHARPY TEST RESULTS UPPER SHELL PLATE M-604-1
5.2-3	CHARPY TEST RESULTS UPPER SHELL PLATE M-604-2
5.2-4	CHARPY TEST RESULTS UPPER SHELL PLATE M-604-3
5.2-5	CHARPY TEST RESULTS INTERMEDIATE SHELL PLATE M-605-1
5.2-6	CHARPY TEST RESULTS INTERMEDIATE SHELL PLATE M-605-2
5.2-7	CHARPY TEST RESULTS INTERMEDIATE SHELL PLATE M-605-3
5.2-8	CHARPY TEST RESULTS LOWER SHELL PLATE M-4116-1
5.2-9	CHARPY TEST RESULTS LOWER SHELL PLATE M-4116-2
5.2-10	CHARPY TEST RESULTS LOWER SHELL PLATE M-4116-3
5.2-11	CHARPY TEST RESULTS BOTTOM HEAD TORUS M-4112-1
5.2-12	CHARPY TEST RESULTS BOTTOM DOME M-4111-1
5.2-13	CHARPY TEST RESULTS CLOSURE HEAD DOME M-4110-1
5.2-14	CHARPY TEST RESULTS CLOSURE HEAD TORUS M-4109-1
5.2-15	CHARPY TEST RESULTS OUTLET NOZZLE SAFE ENDS M-4105-1&2
5.2-16	CHARPY TEST RESULTS INLET NOZZLE SAFE ENDS M-4104-1 Thru 4
5.2-17	CHARPY TEST RESULTS OUTLET NOZZLE M-4103-1

CHAPTER 5

LIST OF FIGURES (Cont'd)

Figure	Title
5.2-18	CHARPY TEST RESULTS OUTLET NOZZLE M-4103-2
5.2-19	CHARPY TEST RESULTS INLET NOZZLE M-4102-1
5.2-20	CHARPY TEST RESULTS INLET NOZZLE M-4102-2
5.2-21	CHARPY TEST RESULTS INLET NOZZLE M-4102-3
5.2-22	CHARPY TEST RESULTS INLET NOZZLE M-4102-4
5.2-23	CHARPY TEST RESULTS CLOSURE HEAD FLANGE M-4101-1
5.2A-1	STEAM GEN. PRESSURE COMPLETE LOSS OF TURBINE - GENERATOR LOAD WITH DELAYED REACTOR TRIP
5.2A-2	OPTIMIZED SAFETY VALVE SIZING
5.2A-3	MAX. REACTOR COOLANT SYST. PRESSURE VS. TIME FOR WORST CASE LOSS OF LOAD INCIDENT WITH DELAYED REACTOR TRIP
5.2A-4	MAX. REACTOR POWER VS. TIME FOR WORST CASE LOSS OF LOAD INCIDENT
5.3-1	TYPICAL SURVEILLANCE CAPSULE ASSEMBLY
5.3-2	TYPICAL CHARPY IMPACT COMPARTMENT ASSEMBLY
5.3-3	TYPICAL TENSILE - MONITOR COMPARTMENT DETAIL
5.3-4	LOCATIONS OF SURVEILLANCE CAPSULE ASSEMBLIES
5.3-5	TRANSITION TEMPERATURE SHIFT FOR ST LUCIE UNIT 2 - PLATE AND WELD METAL
5.3-6	RCS PRESSURE/TEMPERATURE LIMITATIONS 0 TO 10 YEARS OPERATION
5.3-7	RCS PRESSURE/TEMPERATURE LIMITATIONS 10 TO 40 YEARS OPERATION
5.3-8	REACTOR VESSEL
5.4-1	REACTOR COOLANT PUMP
5.4-2	REACTOR COOLANT PUMP PERFORMANCE
5.4-3	REACTOR COOLANT PUMP SEAL AREA
5.4-4	ANTIROTATIONAL DEVICE
5.4-5	MOTOR FLYWHEEL ASSEMBLY

CHAPTER 5

LIST OF FIGURES (Cont'd)

Figure	Title
5.4-6	STEAM GENERATOR
5.4-7	TWO TRAIN COOLDOWN FROM 300 F
5.4-8	ONE TRAIN COOLDOWN FROM 300 F
5.4-9	SHUTDOWN COOLING SUCTION VALVES POWER & CONTROL
5.4-10	PRESSURIZER
5.4-11	PRESSURIZER LEVEL CONTROL
5.4-12	PRESSURIZER PRESSURE CONTROL PROGRAM
5.4-13	QUENCH TANK
5.4-14	PRESSURIZER SAFETY VALVE
5.4-15	POWER-OPERATED RELIEF VALVE

TABLE 5.2-1

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)
Reactor vessel head CEDM nozzles	SB-166
Vessel Internals[a]	Austenitic Stainless Steel and NiCrFe alloy
Fuel Cladding[a]	Zircaloy-4
Instrument nozzles	SB-167
Control element drive mechanism housings	
Lower	SA-182 Type 403 stainless steel Special Code Case 1334 with end fittings to Sb-166 and/or SA-182 Type 348 stainless stee!
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
Pressurizer -	
Shell	SA-533 Grade B Class 1
Cladding[a]	Weld deposited austenitic stainless steel with greater than 5% delta ferrite or NiCrFe alloy (equivalent to SB-168)

(a) Materials exposred to reactor coolant

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

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 nozzle safe ends[a]	SA-182, Type 316
Studs and nuts	SA-540-B24
Steam generator	
Primary head (plate and forging)	SA-533 Grade B, Class 1 (plate) SA-508 Class 2 (forging)
Safe ends	SA-508 Class 1 (forging)
Primary head cladding[a]	Weld deposited austenitic stainless steel with greater than 5% delta ferrite
Tubesheet	SA-508 Class 2 (forging)
Tubesheet stay	SA-508 Class 2 (forging)
Tubesheet cladding[a]	Weld deposited NiCrFe alloy (equivalent to SB-166)
Tube[a]	NiCrFe alloy (SB-163)
Secondary shell and head	SA-533 Grades A, and B Class 1 SA-516 Grade 70
Secondary nozzles	SA-508 Class 1 or Class 2
Secondary nozzle safe ends	SA-508 Class 1
Secondary instrument nozzles	SA-106 Class B
Studs and nuts	SA-540 Grade B24 and SA-193 Grade B7
Sliding base support	A533 Class 2 Grade B Key - A291 Class 3A Charpy V-notch 15 mils internal expansion at 50°F

TABLE 5.2-3 (Cont'd)

Component	Component
Reactor coolant pumps	
Casing[a]	SA-351 Grade CF8M
Internals[a]	Austenitic stainless steel (SA-351 Grade CF8, ASTM A-479 Type 316 ASTM-A-240 Type 316, SA-182 GR F 304)
Studs and Nuts	SA-540 Grade B23 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
Spray Line Fittings	SA-403, Type WP 316 SA-182, F 316 SA-376, TP 316 ASTM-B-366, Grade WNPCI
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

TABLE 5.2-3 (Cont'd)

Component	Component
Reactor coolant piping (Cont'd)	
Spray nozzle safe ends[a]	SA-182 - F316
Letdown and drain or drain nozzle forgings	SA-105 - Grade II
Letdown and drain or drain nozzle safe ends[a]	SA-182 - F316
Sampling or pressure[a] measurement nozzles	SB-166
Sampling or pressure measurement nozzle safe ends[a]	SA-182 - F316
RTD nozzles[a]	SB-166
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 (TP-316), 564 (GR.630) SA-479T (TP-347, 348, 316L)
Pipes	ASTM A-312 (GR TP 304)
Fittings	ASTM A-182 (GR-304) A-403 (GR-WP 304W) A-351 (GR CF8)
Flanges	ASTM A-182 (GRF-304) A-351 (GR-CF8)
Restrictors	ASME SA-182 TP316
Bolts, Nuts	ASTM A-193 GR B7 A-194 GR 2H

5.4 COMPONENT AND SUBSYSTEM DESIGN

5.4.1 REACTOR COOLANT PUMPS

5.4.1.1 Design Bases

The reactor coolant pumps provide sufficient forced circulation flow through the Reactor Coolant System to assure adequate heat removal from the reactor core during power operation. A low limit on the reactor coolant pump flowrate is established to assure that specified fuel design limits are not exceeded. Design flow is derived on the basis of the thermal-hydraulic considerations presented in Section 5.2.

The reactor coolant pump and motor assembly, in conjunction with the flywheel, provide sufficient coast down flow following loss of power to the pumps to assure adequate core cooling.

The reactor coolant pump pressure boundary is designed for the transients given in Subsection 3.9.1.1 so that the ASME Code, Section III allowable stress limits are not exceeded for the specified number of cycles.

The reactor coolant pump parameters and design requirements are listed in Table 5.4-1.

5.4.1.2 Description

The reactor coolant is circulated by four vertical, single bottom suction, horizontal discharge, centrifugal, motor driven pumps as shown in Figure 5.4-1. The design parameters for the pumps are given in Table 5.4-1. The piping and instrumentation diagram for the reactor coolant pump is shown in Figure 5.1-6. The pump performance curve is shown in Figure 5.4-2.

5.4.1.2.1 Reactor Coolant Pump Assembly

The reactor coolant pump assembly consists of the following:

a) Pump Case

b) Rotating Assembly
(Containing the impeller with a welded impeller locknut)

c) Pump Case Cover

d) Motor Mount

e) Motor Assembly

5.4.1.2.1.1 Pump Case and Motor Mount

The reactor coolant pump motor is connected to and supported by the pump case through the motor mount. There are two openings on opposite sides of the motor mounts that provide access for assembly of the flanged rigid coupling between the motor and pump and for seal cartridge replacement.

5.4.1.2.1.2 Rotating Assembly

The pump rotating assembly consists of impeller, water lubricated radial hydrostatic bearing, rotor, seal coolant recirculating impeller and rotating elements of the seal cartridge assembly. The radial bearing, one of three used for pump motor shaft support, is located just above the pump impeller. The upper radial bearing and the axial thrust bearing are located on the motor shaft. The seal cartridge and recirculating impeller are located above the thermal barrier formed by the close clearance between the pump shaft and the pump case cover.

5.4.1.2.1.3 Pump Case Cover

The pump case cover assembly includes the coiled tubing heat exchanger which cools the seal cartridge and thermal barrier, the seal cartridge assembly, the thermal barrier, the radial bearing stator and the upper and lower impeller labyrinth seals.

The seal cartridge consists of four-face type mechanical seals; three full pressure seals mounted in tandem and a fourth low pressure vapor seal designed to withstand system design pressure when the pumps are not operating. A controlled bleed off flow through the seals is used to cool the seals and to equalize the pressure drop across each seal. The controlled bleedoff flow is collected in the volume control tank of the Chemical and Volume Control System. Leakage past the vapor seal is collected in the Waste Management System through drainage to the reactor drain tank.

The seal cartridge assembly is cooled by circulating the controlled leakage through a coiled tube heat exchanger cooled by component cooling water and integral with the pump case cover. The seal coolant recirculation is done by the recirculating impeller located directly below the seal cartridge. The seal cartridge concept reduces the time required for seal maintenance thereby lowering personnel radiation exposure time. The seal cartridge can be removed without draining the pump case. Details of the seal cartridge are shown in Figure 5.4-3.

The seal design life is at least two years. Each seal is designed to accept the full operating pressure of the Reactor Coolant System. The first three seals of the cartridge assembly normally operate with a pressure differential equal to one-third of the operating pressure and with only a slight pressure differential across the vapor seal. The seal rotors are titanium carbide operating against a hard carbon faced stator.

5.4.1.2.1.4 Motor Assembly

The motor assembly includes the following:

a) Air Cooler

b) Motor Bearing Lubrication

c) Oil Lift Pumps

d) Motor Shaft

e) Upper and Lower Radial Guide Bearings

f) Axial Thrust Bearing

g) Flywheel

h) Anti-Reverse Rotation Device

i) Motor

The heat exchanger cooling water is supplied from the Component Cooling Water System. Two 10 hp ac oil lift pumps are used to support the pump-motor shaft assembly during startup and shutdown of the reactor coolant pumps. The motor-pump bearing support system includes a Kingsbury double acting thrust bearing, upper and lower radial bearings in the motor and a radial hydrostatic bearing located above the pump impeller. The piping and instrumentation diagram for the lube oil and cooling system of the pumps is shown on Figure 5.1-6. The flywheel and motor-pump rotating assembly has a minimum total amount of inertia of 100,000 psf to improve pump coastdown characteristics in order to meet system requirements during a loss of pump power condition.

Each pump-motor assembly is equipped with an anti-rotation device shown in Figure 5.4-4 to preclude reverse rotation caused by backflow through the impeller. The device stops the pump when it decelerates from normal speed (900 rpm) to zero speed while the remaining reactor coolant pumps tinue to operate. The anti-reverse device consists of a rotating disc keyed to the motor shaft, and a stationary disc which is bolted to the motor frame. The stationary disc contains several detents each with ramped sides and flats on top of the detents and in the troughs between them. The rotating elements contain several holes in which the retaining pins are located. When reactor coolant pump rotation stops, each pin drops to the flat between detents, and reverse rotation is prevented by the pin which bears against the vertical side of a detent. When motor rotation is started in the normal direction, the pins ride up the ramped sides of the detents and are locked against the sides of the holes in the rotating disc by centrifugal force. No parts are in contact when the motor is operating at rated speed and no lubrication is required for the device. One pin is capable of holding the pump stationary against the torque produced by reverse flow or by the application of 100 percent voltage in reversed phase rotation.

The reactor coolant pump motor is sized for continuous operation at the flows resulting from four-pump operation or partial pump operation with 0.74 specific gravity water. The motor service factor is sufficient to allow 500 heatup cycles. The motors are designed to start and accelerate to speed under full load when 80 percent or more of normal voltage is applied. The motors are contained within NEMA Standard 1-1.20 drip-proof enclosures and are equipped with electrical insulation suitable for a zero to 100 percent humidity and radiation environment of 30R/hr of gamma. The motor cross section is shown in Figure 5.4-5.

5.4.1.3 Evaluation

The reactor coolant pumps are sized to deliver flow that equals or exceeds the design flowrate used in the thermal hydraulic analysis of the Reactor Coolant System. Analysis of steady state and anticipated transients is performed assuming the minimum design flow rate. Tests are performed to evaluate reactor coolant pump performance during the post core load hot functional testing to verify adequate flow.

Leakage from the reactor coolant pump past the pump shaft is controlled by the shaft seal assembly. Reactor coolant entering the seal chambers is cooled and collected in closed systems so that reactor coolant leakage to containment is essentially zero. In the event of a seal malfunction, instrumentation in the form of pressure transducers, a flow meter, and a temperature detector is provided to alert the operator to a potential problem.

Component cooling water to the reactor coolant pumps is not required to ensure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shutdown the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10CFR100. Low component cooling water flow to each pump is indicated and alarmed in the control room. The component cooling water flow from the reactor coolant pumps is sensed by four separate redundant transmitters and low flow is indicated and alarmed in the control room. If the component cooling water flow from the reactor coolant pumps is not restored in 10 minutes, the system automatically trips the reactor and the reactor coolant pumps could be tripped by the operator manually, allowing the system to be cooled down by natural circulation flow. Thermal lag in the reactor coolant pump and motor makes them relatively insensitive to loss of component cooling water flow.

The reactor coolant pumps, by design and field experience, are not susceptible to seal failure resulting from loss of seal cooling water. The reactor coolant pumps are equipped with four series-arranged face seals, all of which are designed for 2500 psid. The ΔP across any one of the three main seals during normal operation is 750 psi. The loss of any single seal would result in a ΔP of approximately 1100 psi. A seal leakage chamber structurally designed for 2500 psia is provided to collect controlled seal leakage and conduct it to a closed system. The fourth face seal is provided as an integral part of the seal leakage chamber to prevent liquid or gaseous leakage from escaping to the atmosphere. This seal is designed to operate normally against a backpressure of 25 to 250 psia and is capable of holding against 2500 psia in the static condition and during coastdown following failure of the three series-arranged main seals. When holding against 2500 psia in the static condition, the seal leakage should not exceed the normal operating seal leakage.

The seals have been specified and tested for 10 minutes of RCP operation without cooling water to the RCP seals without incurring seal damages.

However, four reactor coolant pumps with seals of similar design have been operated for up to 40 minutes with no component cooling water flow. While there was some increase in controlled seal leakage (to the closed system), the mechanical seals were subsequently dismantled and refurbished without finding major damage. Therefore, a loss of component cooling water flow to the reactor coolant pumps for up to 40 minutes is not expected to result in a complete seal failure.

In the event of an actuation of containment isolation and subsequent isolation of CCW, the reactor coolant pumps are tripped, as discussed above, resulting in no requirement for component cooling water.

In the event of a break in the reactor coolant pump suction piping, a high reverse flow through the pump is prevented by the anti-reverse rotation device, as described in Subsection 5.4.1.2.1.4. In the event of a discharge line break, increased flow through the pump tends to accelerate the pump impeller and flywheel in the forward direction. A detailed evaluation of this incident relating to the integrity of the flywheel is presented in Subsection 5.4.1.4.

5.4.1.4 Reactor Coolant Pump Flywheel Integrity

The following design conditions and material specifications for the flywheels are consistent with the recommendations of Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity" October 1971 (R0).

5.4.1.4.1 Flywheel Material Specification

The material used to manufacture the flywheel was produced by a process that minimizes flaws by a commercially acceptable process such as the vacuum melt and degassing process which provides adequate fracture toughness properties. The acceptance criteria for flywheel design is compatible with the safety philosophy of the reactor coolant pressure boundary criteria as appropriate considering the inherent design and functional requirement differences between the pressure boundary and the flywheel.

a) The nil ductility transition temperature (NDTT) of the material, as obtained from the dropweight tests (DWT) performed in accordance with the Specification ASTM E-208-66T was no greater than 10 F.

b) The Charpy V-Notch (Cv) upper shelf energy level, in the "weak" (WR) direction, as obtained per ASTM-A-370 was no less than 50 ft.-lbs. A minimum of three Cv specimens were tested from each plate or forging.

c) The minimum fracture toughness of the material at the normal operating temperature of the flywheel is equivalent to a dynamic stress intensity factor K_{Ic} (dynamic) of at least 100 Ksi$\sqrt{in}$. Compliance was demonstrated by either of the following:

 1) Testing of the actual material of the flywheel to establish the K_{Ic} (dynamic) value at the normal operating temperature.

2) Use of a lower bound fracture toughness curve obtained from tests on the same type of material. The curve was translated along the temperature coordinate until the K_{Ic} (dynamic) value of 45 Ksi$\sqrt{in.}$ is indicated at the NDTT of the material, as obtained from dropweight tests.

d) Each finished flywheel was subjected to a 100 percent volumetric ultrasonic inspection from the flat surface per ASME Code, Section III. This inspection was performed on the flywheel after final machining and overspeed test.

e) The flywheel is flame cut; at least 1/2 in. of stock was left on the outer and bore radii for machining to final dimensions.

f) The flywheel was subjected to a magnetic particle or liquid-penetrant examination per ASME Code, Section III before final assembly. The inspection was performed on finished machine bores, key ways, and on both flat surfaces to a radial distance of eight in. ... beyond the final largest machined bore diameter but not including small drilled holes. There are no stress concentrations such as stamp marks, center punch marks, or drilled or tapped holes within eight in. of the edge of the largest flywheel bore.

5.4.1.6.1.2 Flywheel Design Criteria

The flywheel is designed to withstand normal operating conditions, anticipated transients, and the design basis loss of coolant accident loadings combined with the safe shutdown earthquake loadings.

The following criteria are satisfied:

a) The combined stresses, both centrifugal and interference, at normal operating speed do not exceed 1/3 of the minimum specified yield strength for the material selected in the direction of maximum stress.

b) The design speed of the flywheel is 10 percent above that resulting from a turbine generator overspeed event, or 125 percent of normal operating speed.

c) The combined centrifugal and interference stresses at design speed are limited to 2/3 of the minimum specified yield strength. Design speed is defined as turbine generator overspeed plus 10 percent, or 125 percent of normal operating speed.

d) The motor and pump shaft and bearings can withstand any combination of normal operating loads, anticipated transients, and the design basis loss of coolant accident combined with the safe shutdown earthquake.

e) Each flywheel was tested at design speed, 125 percent of normal operating speed, as defined in 2.b above.

f) The flywheel is accessible for 100 percent inplace volumetric ultrasonic inspection. The flywheel motor assembly is designed to allow such inspection with a minimum of motor disassembly.

5.4.1.5 Reactor Coolant Pump Instrumentation

The reactor coolant pumps and motors are equipped with the instrumentation necessary for proper operation and to warn of incipient failures. A description of the major channels follows (see Figure 5.1-3). Measurement channels are typical for each reactor coolant pump.

5.4.1.5.1 Temperature

5.4.1.5.1.1 Motor Stator Temperature

Each reactor coolant pump motor is provided with six thermocouples embedded in the stator windings. Indication of stator temperature is provided by the plant computer. During initial reactor coolant pump testing, the highest reading thermocouple is selected for this temperature measurement channel. High temperature is detrimental to motor winding insulation life, and may be caused by high ambient temperature, reduction in the cooling air flow to the stator, or inadequate time delay between successive starts of the motor.

5.4.1.5.1.2 Motor Bearing Temperatures

High temperature alarms for the oil lubricated bearings are:

a) Motor upper guide bearing

b) Upper and lower thrust bearings

c) Motor lower guide bearing

High temperature is indicative of bearing or oil supply problems.

5.4.1.5.1.3 Pump Controlled Bleedoff Temperature

The temperature of the controlled bleedoff flow is provided. A high temperature condition is an indication that the seal assembly or seal water cooler is not operating properly.

5.4.1.5.1.4 Seal Water Cooler Component Cooling Water Outlet Temperature

Temperature sensors (TE-1151, 1161, 1171, 1181) are provided at the seal water outlet. A high temperature condition is an indication that the cooler has developed a leak or that the component cooling water flow has decreased. High temperatures are alarmed.

5.4.1.5.2 Pressure

5.4.1.5.2.1 Reactor Coolant Pump Seal Pressures

The middle, upper and vapor seal cavities in each reactor coolant pump are provided with pressure elements that generate a signal proportional to the pressure within the cavity. High and low pressure alarms for the upper seal cavity and the vapor seal cavity are provided. Pressure indication of all three is also provided.

5.5.1.5.2.2 High Pressure Oil Lift Pump Discharge Pressures

Pressure switches at each high pressure oil lift pump discharge actuate indicating lights and alarms on low pressure in the control room. In the event of a failure of one of the oil lift pumps, the second oil lift pump must be started. A separate measurement channel provides a control signal to the respective reactor coolant pump circuit; which prevents the starting of the reactor coolant pump if insufficient oil lift pressure exists. Another separate measurement channel provides local indication of pressure in the oil lift pump discharge header.

5.4.1.5.2.3 Reactor Coolant Pump Differential Pressure

Two independent differential pressure transmitters are provided on each reactor coolant pump. The differential pressure signal is indicated in the control room. A calibration curve is used to relate pump differential pressure to pump flow.

5.4.1.5.2.4 Casing Main Closure Gasket Leakage Pressure

A pressure indicator and pressure switch are provided on each reactor coolant pump to monitor the pressure between the double casing main closure gaskets. High pressure in the cavity between the gaskets indicates leakage of the inboard gaskets and is alarmed in the control room.

5.4.1.5.3 Flow

5.4.1.5.3.1 Reverse Rotation Indicator Switch

A flow switch in the lube oil system mounted near the main thrust bearing bracket provides an indication that the reactor coolant pump motor is turning in the reverse direction. This switch causes an alarm in the control room.

5.4.1.5.3.2 Pump Controlled Bleedoff Flow

A flowmeter is used to measure the controlled bleedoff flow from the bleedoff seal cavity to the CVCS. This instrument provides an indication of the flowrate and annunciates high and low flow alarms.

5.4.1.5.3.3 Motor Circulating Oil System Flow

A lube oil flow switch is provided at the inlet to the lube oil cooler. Should the lube oil flow to the cooler fall below a predetermined setpoint, a low flow alarm is actuated.

5.4.1.5.4 Level

A level sensing transmitter in each oil reservoir transmits a signal for level indication and high and low alarms.

5.4.1.5.5 Vibration

Motor vibration is sensed by a vibration switch attached to the pump motor casing. Excessive vibration is alarmed.

5.4.1.6 Testing and Inspection

The reactor coolant pressure boundary is nondestructively inspected as required by ASME Code, Section III for Code Class 1 components. The reactor coolant pump casing inspections include complete radiography and liquid penetrant or ultrasonic testing. The reactor coolant pump receives a hydrostatic pressure test in the vendor's shop and with the Reactor Coolant System. In-service inspection of the reactor coolant pump pressure boundary is performed during plant life in accordance with ASME Code, Section XI.

The reactor coolant pump assembly is performance tested in the vendor's shop over at least the normal operating range in accordance with the Standards of the Hydraulic Institute. Tests also demonstrate the ability of the reactor coolant pump to function under various operating conditions specified. Tests commonly performed are hot and cold performance and start-stop cycling. Vibrations are monitored at several places on the reactor coolant pump during shop testing.

The reactor coolant pump motors undergo a "routine" test in accordance with NEMA MG-1. This test also confirms that the motors are within their vibration limits. Each motor is tested further by being used as the driver for the reactor coolant pump assemblies during the pump manufacturer's shop testing.

To the greatest extent practicable, all conditions of normal operation of the reactor coolant pumps are duplicated during testing.

The reactor coolant pump flywheel inspections and testing are described in Subsection 5.4.1.4.

5.4.2 STEAM GENERATORS

5.4.2.1 Design Bases

The two steam generators are designed to transfer 2570 MWt from the Reactor Coolant System to the Main Steam System, producing approximately 11.206×10^6 lb/h of 815 psia saturated steam, when provided with 435 F feedwater. Moisture separators and steam driers in the shell side of the steam generator limit the moisture content of the steam to 0.20 wt% during normal operation at full power. The steam generator design parameters are listed in Table 5.4-2. The steam generators, including the tubes, are designed for the Reactor Coolant System transients listed in Subsection 3.9.1.1 so that the code allowable stress limits are not exceeded for the specified number of cycles. All transients have been established based on conservative assumptions of operating conditions in consideration of supportive system design capabilities. The steam generators are capable of sustaining the following additional design transients without exceeding code allowable stress limits:

a) Ten secondary side hydrostatic tests with secondary side pressurized to 1250 psia with the primary side at atmospheric pressure. The minimum shell side temperature for this test is 100 F.

b) Two hundred secondary side leak tests with the secondary side pressurized from 920 psia to design pressure, with the primary side pressurized so that the tube differential pressure (secondary to primary) does not exceed 820 psi (test condition). The secondary side temperature shall be 100-200 F.

c) Fifteen thousand cycles of adding 40 F feedwater at 600 gpm to each of the steam generators through the main feedwater nozzle when at hot standby conditions (normal condition). The basis is nominal operating conditions assuming intermittent feeding of the steam generators.

d) Eight cycles of adding 40 feedwater at 650 gpm to each of the steam generators after a loss of normal feedwater. This feedwater flow may be introduced while the secondary side is dry: at 610 F and atmospheric pressure.

e) Four thousand pressure transients of 85 psi across the primary divider plate in either direction caused by starting and stopping reactor coolant pumps (normal condition).

5.4.2.1.1 Steam Generator Materials

The pressure boundary materials used in the construction of the steam generator are listed in Table 5.2-3. These materials are in accordance with the ASME Code, Section III plus code case interpretations as specified in Subsection 5.2.1.

The Code Class 1 components of the steam generator meet the fracture toughness requirements of the ASME Code and 10CFR50, Appendix G as discussed in 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 it 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 0.048 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 600 composition.

The steam generator contains 8411 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 of the egg crate type.

Feedwater enters the steam generator through the feedwater nozzle where it is distributed via a feedwater distribution ring. The feedwater ring is constructed with discharge nozzles which are configured in the form of a "C". These nozzles are welded to the top of the ring and attached to the spargers in the bottom of the ring. These "C" tubes 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.

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 existing 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 centrifugal-type separators. These impart a centrifugal motion to 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 corrugated plate 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 to the concrete structure.

The steam generators are located at a higher elevation than the reactor vessel. The elevation difference creates 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 generators are tubed with 0.750 inch OD by .048 wall tubes. The tubes are fabricated from Inconel 600 to insure compatibility with both the primary and secondary waters. The design incorporates a general corrosion allowance that provides for reliable operation over the plant design lifetime.

Localized corrosion has led to steam generator tube leakage in some operating reactor 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 Wall Thinning

The design steady state and transient conditions specified in the design of the steam generator tubes are discussed in Subsection 1.9.1.1. At least 0.012 inches of excess material is intentionally

provided to accommodate degradation of tubes due to normal uniform corrosion that may occur during the service lifetime.

b) Denting

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 (ie, 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. When 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 be concentrated.

2) The bulk water being concentrated must have condenser leakage impurities that produce 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 an antivibration support system that does not use drilled support plates. Supports of the same type, "egg crates", have been utilized to some extent in all CE supplied commercial steam generators within the United States.

The egg crate system reduces susceptibility to tube denting by providing larger clearances and increased flow area around the tubes, so that the clearances between the tubes and their supports are less likely to become plugged by corrosion products.

St. Lucie Unit 2 has a full egg crate support system (all support plates have been eliminated).

c) Potential Effects of Tube Rupture

The steam generator tube rupture incident is a penetration of the barrier between the RCS and the Main Steam System. The integrity of this barrier is significant from the standpoint of radiological safety in that a leaking steam generator tube allows the transfer of reactor coolant into the Main Steam System. Radioactivity contained in the reactor coolant would mix with water in the shell side of the affected steam generator. This radioactivity would be transported by steam to the turbine and then to the condenser or directly to the condenser via the Steam Dump and Bypass System. Noncondensible radioactive gases in the condenser are removed by the Main Condenser Evacuation System and discharged to the plant vent. Analysis of a steam generator tube rupture incident, assuming complete severance of a tube, is presented in Section 15.6.

Experience with nuclear steam generators indicates that the probability of complete severance of a tube is remote. The material used to fabricate the vertical U-tube is a Ni-Cr-Fe alloy. A double-ended rupture has never occurred in a steam generator of this design. The more probable modes of failure, which result in smaller penetrations, are those involving the occurrence of pinholes or small cracks in the tubes, and of cracks in the seal welds between the tubes and tube sheet. Detection and control of steam generator tube leakage is described in Subsection 5.2.5.

d) Composition of Secondary Fluid and Radiological Considerations

Radioactivity connection in the secondary side of the steam generator is dependent upon the activity level of the Reactor Coolant System, the primary to secondary leak rate, and the operation of the Steam Generator Blowdown System. An evaluation of shell side radioactivity concentration is given in Section 11.1.

The recirculation water within the steam generators contains volatile additives necessary for proper chemistry control. These and other chemistry considerations of the Main Steam System are discussed in Subsection 10.3.5.

Materials used in fabrication of the steam generator are not affected by the radiation levels and doses resulting from operation. Although radiation levels are significant for any internal maintenance operations, procedures and equipment have been developed to minimize individual personnel exposure during these operations by allowing rapid completion of individual maintenance operations.

5.4.2.2 Steam Generator In-service Inspection

a) The preservice and in-service inspection programs are developed to comply with the ASME Code, Section XI requirements as appropriate, to permit examinations of the steam generator Code Class 1 and 2 component parts, including the steam generator tubes (refer to Subsection 5.2.4 and Section 6.6).

b) The in-service inspection program of the steam generator tubes is developed to comply with Appendix IV of the ASME Code, Section XI.

1) The program parameters comply with the guidelines recommended in Regulatory Guide 1.83, "In-service Inspection of Pressurized Water Reactor Steam Generator Tubes", July 1975 (R1).

2) The program examination method, equipment and reporting requirements comply to Appendix IV of the ASME Code, Section XI. The program parameters governing the criteria used for tube inspection, inspection intervals, and acceptance criteria (including plugging limits) are included in Technical Specifications.

5.4.3 REACTOR COOLANT PIPING

5.4.3.1 Design Basis

The reactor coolant piping is designed and analysed for normal operation and all transients discussed in Subsection 3.9.1. Loading combinations and stress criteria associated with faulted conditions are presented in Subsection 3.9.1. In addition, certain nozzles are subjected to local transients that are included in the design and analysis of the areas affected. Thermal sleeves are installed in the surge nozzle, safety injection nozzles, and charging nozzles to accommodate these additional transients. Principal parameters are listed in Table 5.4-3. The ASME Code and Addenda the piping is designed to is specified in Subsection 5.2.1.

In addition to being specified as seismic Category I, the following additional vibratory requirement is specified in the engineering specification. The various piping assemblies are designed so that no damage to the equipment is caused by the frequency ranges of 14 to 15 Hz and 70 to 75 Hz. The frequency ranges account for mechanical vibratory excitation of the reactor coolant pump and impeller vane passing pressure variations.

5.4.3.2 Description

Each of the two heat transfer loops contains five sections of pipe; one 42 inch internal diameter pipe between the reactor vessel outlet nozzle and steam generator inlet nozzle, two 30 inch internal diameter pipes from the steam generator's two outlet nozzles to the two reactor coolant pump suction nozzles, and two 30 inch internal diameter pipes from the reactor coolant pump discharge nozzles to the reactor vessel inlet nozzles. These pipes are referred to as the hot leg, the suction legs, and the cold legs, respectively. The other major section of reactor coolant piping is the surge line, a 12 inch schedule 160 pipe between the pressurizer and the hot leg in Loop 2B, and the spray line, a 4 inch Schedule 160 pipe at the pressurizer reduced to two 3 inch schedule 160 pipes between the 4 inch pipe and each cold leg in Loops 2B1 and 2B2. Arrangement of this piping is further described in Subsection 5.1.1.