August 1, 1990

Docket No. 50-389

DISTRIBUTION
See attached sheet

Mr. J. H. Goldberg
Executive Vice President
Florida Power and Light Company
P.O. Box 14000
Juno Beach, Florida 33408-0420

Dear Mr. Goldberg:

SUBJECT: ST. LUCIE UNIT 2 - ISSUANCE OF AMENDMENT RE: PRESSURE/TEMPERATURE LIMITS AND LOW TEMPERATURE OVERPRESSURE PROTECTION (TAC NO. 76016)

The Commission has issued the enclosed Amendment No. 46 to Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your application dated February 7, 1990, as supplemented June 19, 1990.

This amendment incorporates revised pressure/temperature (P/T) limits and the results of a revised low temperature overpressure protection (LTOP) analysis into the Technical Specifications for St. Lucie Unit 2 for up to 15 effective full power years (EFPY) of operation. The current St. Lucie Unit 2 Technical Specifications for P/T and LTOP are applicable to 6 EFPY of operation. Accordingly, the St. Lucie Unit 2 Technical Specifications require revision prior to the plant reaching 6 EFPY.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by

Jan A. Norris, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 46 to NPF-16
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	LA:PD22	PM:PD22	D:PD22	OGC		
NAME	DMiller	JNorris	HBerkow	R. Tell		
DATE	7/11/90	7/18/90	8/1/90	7/24/90		

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Document Name: ST LUCIE AMENDMENT 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

August 1, 1990

Docket No. 50-389

Mr. J. H. Goldberg
Executive Vice President
Florida Power and Light Company
P.O. Box 14000
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Jan A. Norris, Senior Project Manager
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1. Amendment No. 46 to NPF-16
2. Safety Evaluation

cc w/enclosures:
See next page

DATED: August 1, 1990

AMENDMENT NO. 46 TO FACILITY OPERATING LICENSE NO. DPR-67 - ST. LUCIE, UNIT 1

Docket File
NRC & Local PDRs
PDII-2 Reading
S. Varga, 14/E/4
G. Lainas, 14/H/3
H. Berkow
D. Miller
J. Norris
OGC-WF
D. Hagan, 3302 MNBB
E. Jordan, 3302 MNBB
B. Grimes, 9/A/2
G. Hill (4), P-137
Wanda Jones, P-130A
J. Calvo, 11/F/23
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M. Sinkule, R-II
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000102

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Florida Power & Light Company

St. Lucie Plant

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-335

ST. LUCIE PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 46
License No. DPR-67

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Florida Power & Light Company, (the licensee) dated February 7, 1990, as supplemented June 19, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, Facility Operating License No. DPR-67 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and by amending paragraph 2.C.(2) to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 46 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: August 1, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 46

TO FACILITY OPERATING LICENSE NO. NPF-16

DOCKET NO. 50-389

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages	Insert Pages
1-4	1-4
3/4 4-3	3/4 4-3
3/4 4-5	3/4 4-5
3/4 4-10	3/4 4-10
3/4 4-29	3/4 4-29
3/4 4-31a	3/4 4-31a
3/4 4-31b	3/4 4-31b
3/4 4-32	3/4 4-32
3/4 4-35	3/4 4-35
3/4 4-37a	3/4 4-37a
B 3/4 4-1	B 3/4 4-1
B 3/4 4-3	B 3/4 4-3
B 3/4 4-8	B 3/4 4-8
B 3/4 4-11	B 3/4 4-11

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, " Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.12 The ENGINEERED SAFETY FEATURES RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE TREATMENT SYSTEM

1.14 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured, and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the secondary system.

LOW TEMPERATURE OVERPRESSURE PROTECTION RANGE - RCS

1.16 The LOW TEMPERATURE OVERPRESSURE PROTECTION RANGE is that operating condition when (1) the RCS cold leg temperature is less than or equal to that specified in Table 3.4-3, and (2) the Reactor Coolant System is not vented to containment by an opening of at least 3.58 square inches.

MEMBER(S) OF THE PUBLIC

1.17 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.18 The OFFSITE DOSE CALCULATION MANUAL shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and shall include the Radiological Environmental Monitoring Sample point locations.

OPERABLE - OPERABILITY

1.19 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.20 An OPERATIONAL MODE (i.e. MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.21 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loop(s)/train(s) listed below shall be OPERABLE and at least one Reactor Coolant and/or shutdown cooling loops shall be in operation.*

a. Reactor Coolant Loop 2A and its associated steam generator and at least one associated Reactor Coolant pump,**

b. Reactor Coolant Loop 2B and its associated steam generator and at least one associated Reactor Coolant pump,**

c. Shutdown Cooling Train 2A,

d. Shutdown Cooling Train 2B.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required Reactor Coolant and/or shutdown cooling loops OPERABLE, immediately initiate corective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is a shutdown cooling loop, be in COLD SHUTDOWN within 30 hours.

b. With no Reactor Coolant or shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

*All Reactor Coolant pumps and shutdown cooling pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**A Reactor Coolant pump shall not be started with two idle loops and one or more of the Reactor Coolant System cold leg temperatures less than or equal to that specified in Table 3.4-3 unless the secondary water temperature of each steam generator is less than 40°F above each of the Reactor Coolant System cold leg temperatures.

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required Reactor Coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be $\geq$ 10% indicated narrow range level at least once per 12 hours.

4.4.1.3.3 At least one Reactor Coolant or shutdown cooling loop shall be verified to be in operation and circulating Reactor Coolant at least once per 12 hours.

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one shutdown cooling loop shall be OPERABLE and in operation*, and either:

a. One additional shutdown cooling loop shall be OPERABLE#, or

b. The secondary side water level of at least two steam generators shall be greater than 10% indicated narrow range level.

APPLICABILITY: MODE 5 with Reactor Coolant loops filled##.

ACTION:

a. With one of the shutdown cooling loops inoperable and with less than the required steam generator level, immediately initiate corrective action to return the inoperable shutdown cooling loop to OPERABLE status or to restore the required steam generator level as soon as possible.

b. With no shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of at least two steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

---

*The shutdown cooling pump may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

#One shutdown cooling loop may be inoperable for up to 2 hours for surveillance testing provided the other shutdown cooling loop is OPERABLE and in operation.

##A Reactor Coolant pump shall not be started with two idle loops unless the secondary water temperature of each steam generator is less than 40°F above each of the Reactor Coolant System cold leg temperatures.

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two shutdown cooling loops shall be OPERABLE# and at least one shutdown cooling loop shall be in operation.*

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

a. With less than the above required loops OPERABLE, within 1 hour initiate corrective action to return the required loops to OPERABLE status as soon as possible.

b. With no shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and within 1 hour initiate corrective action to return the required shutdown cooling loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.2 At least one shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

#One shutdown cooling loop may be inoperable for up to 2 hours for surveillance testing provided the other shutdown cooling loop is OPERABLE and in operation.

*The shutdown cooling pump may be deenergized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a minimum water level of greater than or equal to 27% indicated level and a maximum water level of less than or equal to 68% indicated level and at least two groups of pressurizer heaters capable of being powered from 1E buses each having a nominal capacity of at least 150 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one group of the above required pressurizer heaters inoperable, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water volume shall be determined to be within its limits at least once per 12 hours.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified to be at least 150 kW at least once per 92 days.

4.4.3.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by verifying that on an Engineered Safety Features Actuation test signal concurrent with a loss of offsite power:

a. the pressurizer heaters are automatically shed from the emergency power sources, and

b. the pressurizer heaters can be reconnected to their respective buses manually from the control room after resetting of the ESFAS test signal.

3/4.4.4 PORV BLOCK VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 Each Power Operated Relief Valve (PORV) Block valve shall be OPERABLE. No more than one block valve shall be open at any one time.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more block valve(s) inoperable, within 1 hour either restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both block valves open, close one block valve within 1 hour, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. The provisions of specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of Action a. or b. above.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2, 3.4-3 and 3.4-4 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} to less than 200°F within the next 30 hours in accordance with Figures 3.4-3 and 3.4-4.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR 50 Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2, 3.4-3 and 3.4-4.

FIGURE 3.4-2
ST. LUCIE-2 P/T LIMITS, 15 EFPY
HEATUP AND CORE CRITICAL

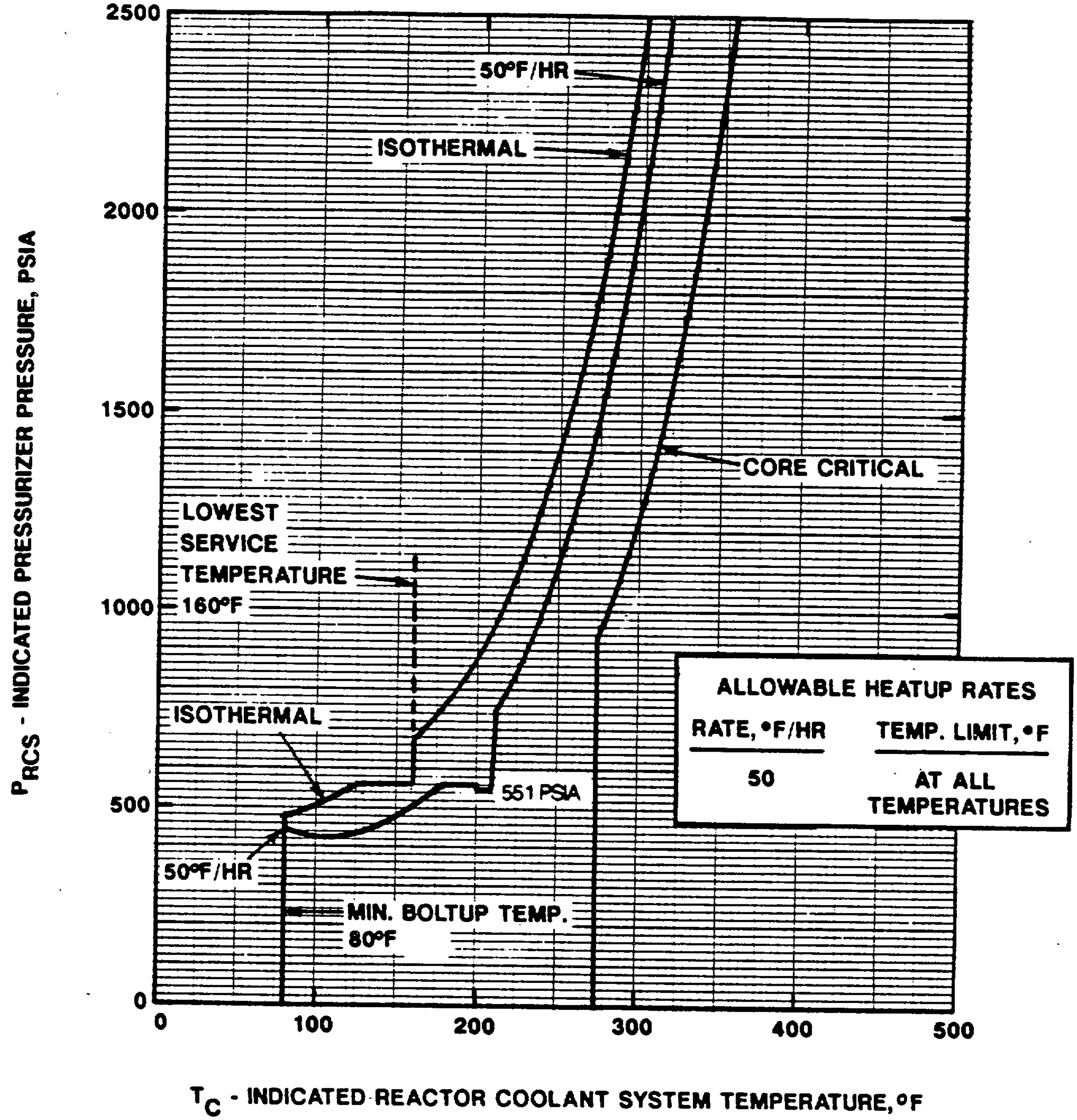


FIGURE 3.4-3
ST. LUCIE-2 P/T LIMITS, 15 EFPY
COOLDOWN AND INSERVICE-TEST

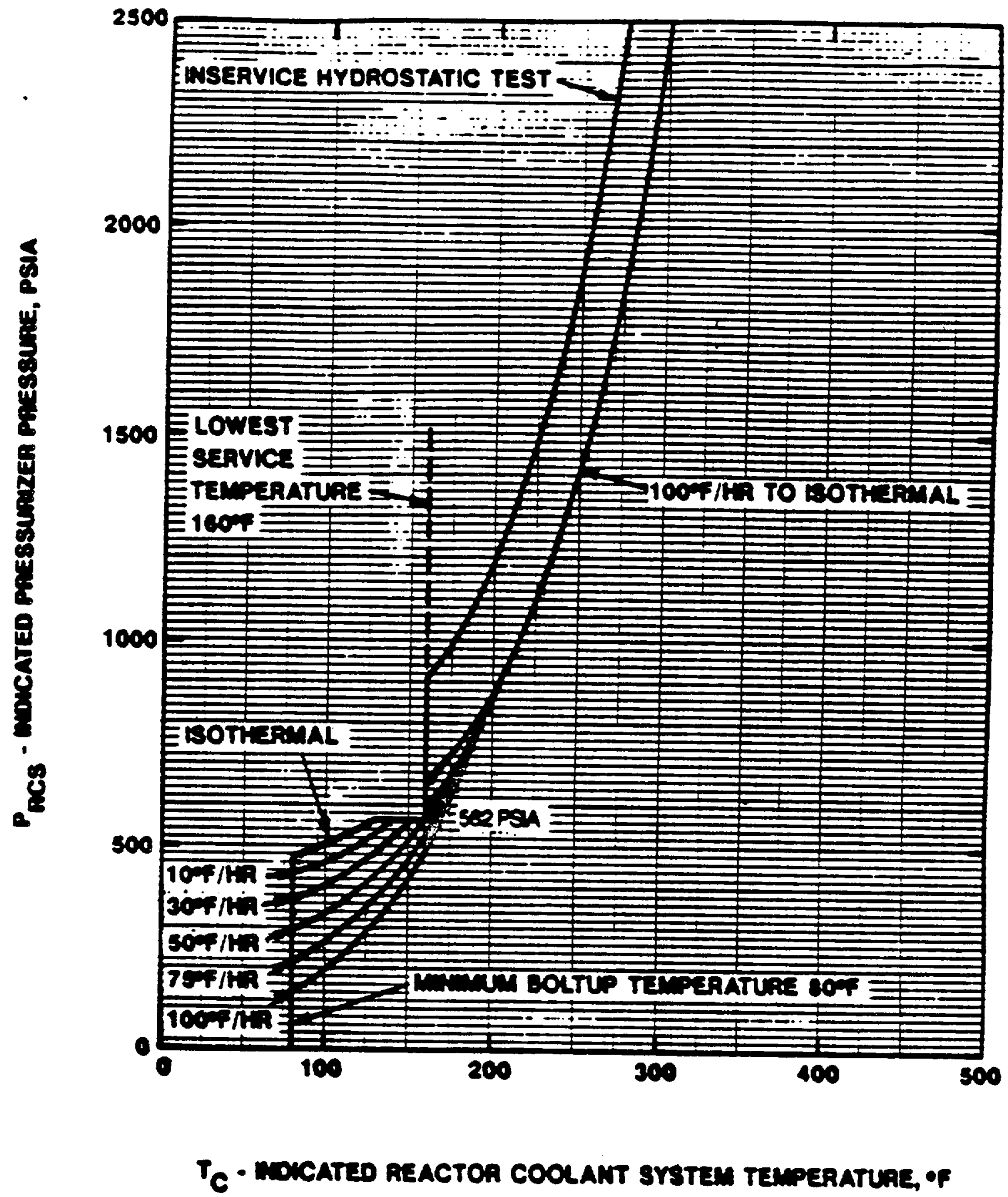
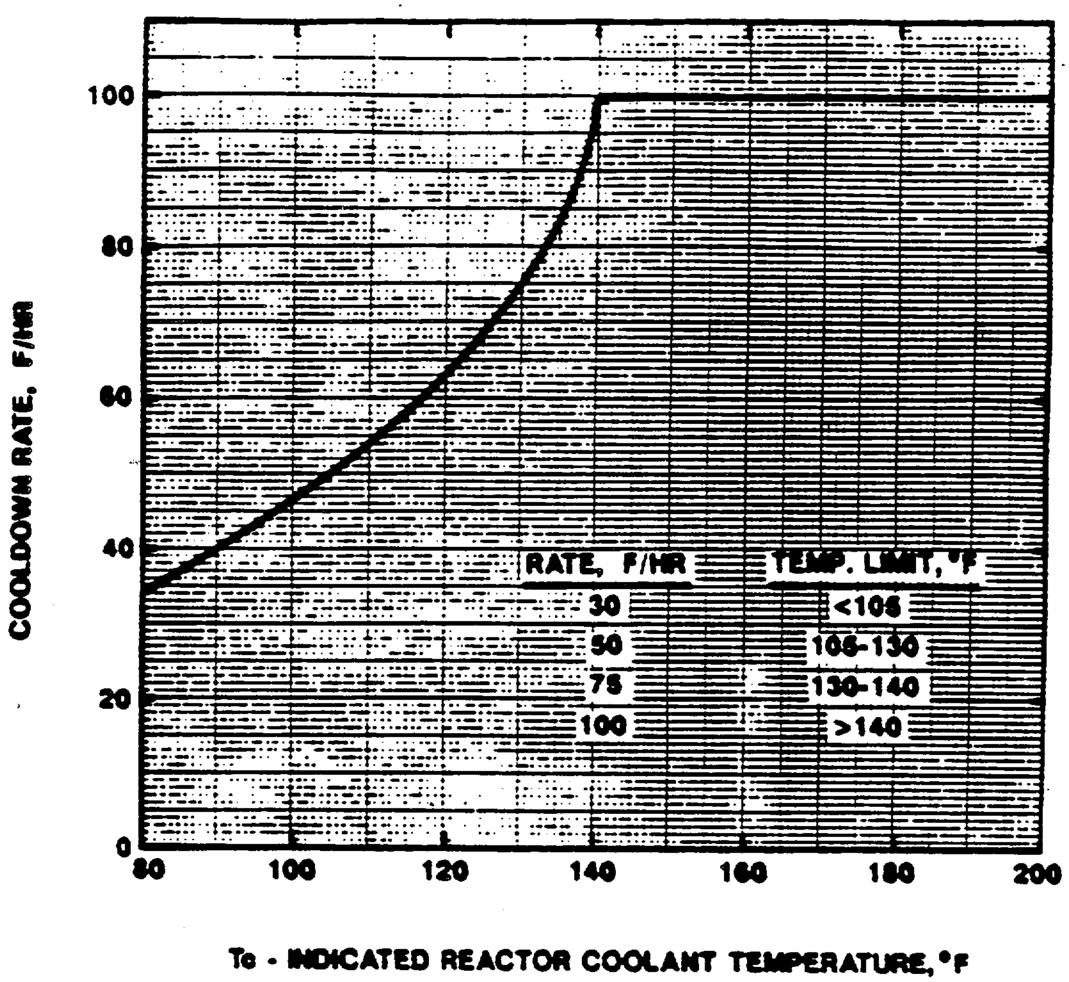


FIGURE 3.4-4
ST. LUCIE-2 P/T LIMITS, 15 EFPY
MAXIMUM ALLOWABLE COOLDOWN RATES



NOTE: A MAXIMUM COOLDOWN RATE OF 100 F/HR IS ALLOWED AT ANY TEMPERATURE ABOVE 140°F

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 Unless the RCS is depressurized and vented by at least 3.58 square inches, at least one of the following overpressure protection systems shall be OPERABLE:

a. Two power-operated relief valves (PORVs) with a lift setting of less than or equal to 470 psia and with their associated block valves open. These valves may only be used to satisfy low temperature overpressure protection (LTOP) when the RCS cold leg temperature is greater than the temperature listed in Table 3.4-4.

b. Two shutdown cooling relief valves (SDCRVs) with a lift setting of less than or equal to 350 psia.

c. One PORV with a lift setting of less than or equal to 470 psia and with its associated block valve open in conjunction with the use of one SDCRV with a lift setting of less than or equal to 350 psia. This combination may only be used to satisfy LTOP when the RCS cold leg temperature is greater than the temperature listed in Table 3.4-4.

APPLICABILITY: MODES 4#, 5 and 6.

ACTION:

a. With either a PORV or an SDCRV being used for LTOP inoperable, restore at least two overpressure protection devices to OPERABLE status within 7 days or:

   1. Depressurize and vent the RCS with a minimum vent area of 3.58 square inches within the next 8 hours; OR

   2. Be at a temperature above the LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE of Table 3.4-3 within the next 8 hours.

b. With none of the overpressure protection devices being used for LTOP OPERABLE, within the next eight hours either:

   1. Restore at least one overpressure protection device to OPERABLE status or vent the RCS; OR

   2. Be at a temperature above the LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE of Table 3.4-3.

#With cold leg temperature within the LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE of Table 3.4-3.

LIMITING CONDITION FOR OPERATION

ACTION (Continued):

c. In the event either the PORVs, SDCRVs or the RCS vent(s) are used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, SDCRVs or vent(s) on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. In addition to the requirements of Specification 4.0.5, operating the PORV through one complete cycle of full travel at least once per 18 months.

SURVEILLANCE REQUIREMENTS (Continued)

b. Performance of a CHANNEL FUNCTIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE.

c. Performance of a CHANNEL CALIBRATION on the PORV actuation channel, at least once per 18 months.

d. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

TABLE 3.4-3

LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE

Operating Period, EFPY	Cold Leg Temperature, F° During Heatup	Cold Leg Temperature, F° During Cooldown
6 < operating period ≤ 15	≤ 247	≤ 230

TABLE 3.4-4

MINIMUM COLD LEG TEMPERATURE FOR PORV USE FOR LTOP

Operating Period EFPY	T_{cold}, F° During Heatup	T_{cold}, F° During Cooldown
6 < operating period ≤ 15	165	165

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.20 during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either shutdown cooling or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two shutdown cooling loops be OPERABLE.

The operation of one reactor coolant pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

The restriction on starting a reactor coolant pump in MODES 4 and 5, with two idle loops and one or more RCS cold leg temperatures less than or equal to that specified in Table 3.4-3 is provided to prevent RCS pressure transients, caused by energy additions from the secondary system from exceeding the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients by (1) sizing each PORV to mitigate the pressure transient of an inadvertent safety injection actuation in a water-solid RCS with pressurizer heaters energized, (2) restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 40°F above each of the RCS cold leg temperatures, (3) using SDCRVs to mitigate RCP start transients and the transients caused by inadvertent SIAS actuation and charging water, and (4) rendering one HPSI pump inoperable when the RCS is at low temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 212,182 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

SAFETY VALVES (Continued)

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the system pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the pressurizer power-operated relief valve or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.3 PRESSURIZER

An OPERABLE pressurizer provides pressure control for the Reactor Coolant System during operations with both forced reactor coolant flow and with natural circulation flow. The minimum water level in the pressurizer assures the pressurizer heaters, which are required to achieve and maintain pressure control, remain covered with water to prevent failure, which could occur if the heaters were energized uncovered. The maximum water level in the pressurizer ensures that this parameter is maintained within the envelope of operation assumed in the safety analysis. The maximum water level also ensures that the RCS is not a hydraulically solid system and that a steam bubble will be provided to accommodate pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The requirement to verify that on an Engineered Safety Features Actuation test signal concurrent with a loss of offsite power the pressurizer heaters are automatically shed from the emergency power sources is to ensure that the non-Class 1E heaters do not reduce the reliability of or overload the emergency power source. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability to control Reactor Coolant System pressure and establish and maintain natural circulation.

3/4.4.4 PORV BLOCK VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs in conjunction with a reactor trip on a Pressurizer Pressure-High signal minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. The opening of the PORVs fulfills no safety-related function and no credit is taken for their operation in the safety analysis for MODE 1, 2, or 3.

Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable. Since it is impractical and undesirable to actually open the PORVs to demonstate their reclosing, it becomes necessary to verify OPERABILITY of the PORV block valves to ensure the capability to isolate a malfunctioning PORV. As the PORVs are pilot operated and require some system pressure to operate, it is impractical to test them with the block valve closed.

The PORVs are sized to provide low temperature overpressure protection (LTOP). Since both PORVs must be OPERABLE when used for LTOP, both block valves will be open during operation within the LTOP range. As the PORV capacity required to perform the LTOP function is excessive for operation in MODE 1, 2, or 3, it is necessary that the operation of more than one PORV be precluded during these MODES. Thus, one block valve must be shut during MODES 1, 2, and 3.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice Inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion.

STEAM GENERATORS (Continued)

Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system primary-to-secondary leakage = 1.0 gpm from both steam generators. Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 0.5 gpm per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of

SPECIFIC ACTIVITY (Continued)

DELETED

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 5.2 of the FSAR.. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients through the reactor vessel wall produce thermal stresses which are compressive at the reactor vessel inside surface and which are tensile at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the outside surface location. However, since neutron irradiation damage is larger at the inside surface location when compared to the outside surface, the inside surface flaw may be more limiting. Consequently, for the heatup analysis both the inside and outside surface flaw locations must be analyzed for the specific pressure and thermal loadings to determine which is more limiting.

During cooldown, the thermal gradients through the reactor vessel wall produce thermal stresses which are tensile at the reactor vessel inside surface and which are compressive at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the inside surface location. Since the neutron indication damage is also greatest at the inside surface location the inside surface flaw is the limiting location. Consequently, only the inside surface flaw must be evaluated for the cooldown analysis.

The heatup and cooldown limit curves Figures 3.4-2, 3.4-3 and 3.4-4 are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate of up to 50 degress F per hour or cooldown rate of up to 100 degrees F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at 15 EFPY, and they include adjustments for pressure differences between the reactor vessel beltline and pressurizer instrument taps.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation will cause an increase in the RT_{NDT}. An adjusted reference temperature can be predicted using a) the initial RT_{NDT}, b) the fluence (E greater than 1 MeV), including appropriate adjustments for neutron attenuation and neutron energy spectrum variations through the wall thickness, c) the copper and nickel contents of the material, and d) the transition temperature shift from the curve shown in Figure B 3/4.4-1 as recommended by Regulatory Guide 1.99, Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves Figures 3.4-2, 3.4-3 and 3.4-4 include predicted adjustments for this shift in RT_{NDT} at 15 EFPY.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10 CFR Appendix H, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the delta RT_{NDT} determined from the surveillance capsule is different from the calculated delta RT_{NDT} for the equivalent capsule radiation exposure. The lead factors shown in Table 4.4-5 are the ratio of neutron flux at the surveillance capsule to that at the reactor inside surface.

The pressure-temperature limit lines shown on Figures 3.4-2, 3.4-3 and 3.4-4 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The maximum RT_{NDT} for all Reactor Coolant System pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 60°F. The Lowest Service Temperature limit line shown on Figures 3.4-2, 3.4-3 and 3.4-4 is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps, and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs, two SDCRVs or an RCS vent opening of greater than 3.58 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold leg temperatures are less than or equal to the LTOP temperatures. The Low Temperature Overpressure Protection System has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) a safety injection actuation in a water-solid RCS with the pressurizer heaters energized or (2) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 40°F above the RCS cold leg temperatures with the pressurizer water-solid.

3/4.4.10 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one Reactor Coolant System vent path from the reactor vessel head and the pressurizer steam space ensures the capability exists to perform this function.

The redundancy design of the Reactor Coolant System vent systems serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vent system are consistent with the requirements of Item II.b.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.4.11 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a (g) (6) (i).

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1971 Edition and Addenda through Summer 1973.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 46

TO FACILITY OPERATING LICENSE NO. NPF-16

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT, UNIT NO. 2

DOCKET NO. 50-389

INTRODUCTION

By application dated February 7, 1990, as supplemented June 19, 1990, the Florida Power and Light Company (the licensee) requested an amendment which would incorporate revised pressure/temperature (P/T) limits and the results of a revised low temperature overpressure protection (LTOP) analysis into the Technical Specifications (TS) for St. Lucie Unit 2. The current St. Lucie Unit 2 TS for P/T and LTOP are applicable to 6 effective full power years (EFPY). Accordingly, the St. Lucie 2 TS require revision prior to the plant reaching 6 EFPY. Below is the staff's evaluation of the proposed changes.

The licensee's letter dated June 19, 1990 corrected a typographical error on proposed TS page 3/4 4-31a. This information did not change the staff's initial determination of no significant hazards consideration as published in the Federal Register on March 7, 1990 (55 FR 8224).

PRESSURE/TEMPERATURE LIMITS

In response to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," the licensee requested permission to revise the P/T limits in the St. Lucie 2 TS, Section 3.4. This revision also changes the effectiveness of the P/T limits from 6 to 15 EFPY. The proposed P/T limits were developed using Regulatory Guide (RG) 1.99, Revision 2. The proposed revision provides up-to-date P/T limits for the operation of the Reactor Coolant System (RCS) during heatup, cooldown, criticality, and hydrotest.

To evaluate the P/T limits, the staff uses the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Rev. 2; Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide TS for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the TS. The P/T limits are among the limiting conditions of operation in the

TS for all commercial nuclear plants in the U.S. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees and permittees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

Evaluation

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the St. Lucie 2 reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the material with the highest ART at 15 EFPY was the intermediate shell plate M-605-1 with 0.11% copper (Cu), 0.61% nickel (Ni) and an initial RT_{NDT} of 30°F.

The licensee has removed one surveillance capsule from St. Lucie 2. The results from capsule W-83 were published in Babcox & Wilcox Report BAW-1880. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material, intermediate shell plate M-605-1, the staff calculated the ART to be 139.9°F at 1/4T (T = reactor vessel beltline thickness) and 118.6°F for 3/4T at 15 EFPY. The staff used a neutron fluence of 1.09E19 n/cm² at 1/4T and 3.86E18 n/cm² at 3/4T. The ART was determined by Section 1 of RG 1.99, Rev. 2, because only one surveillance capsule has been removed from the St. Lucie 2 reactor vessel.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 140°F at 15 EFPY at 1/4T for the same limiting intermediate shell plate material. Substituting the ART of 140°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange reference temperature of 0°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. The licensee has unirradiated USE data for all plate materials, but does not have unirradiated USE data for any of the beltline welds. The staff has determined that all the plate materials for which data are available will meet the requirement that the Charpy USE at end of life be above 50 ft-lb. Since the unirradiated USEs of the welds are unavailable, the staff will monitor closely the irradiated USEs of the welds to ascertain that the USEs will be above 50 ft-lb.

Technical Finding

The staff concludes that the proposed P/T limits for the RCS for heatup, cooldown, leak test, and criticality are valid through 15 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The licensee's submittal also satisfies Generic Letter 88-11 because the licensee used the method in RG 1.99, Rev. 2 to calculate the ART. Hence, the proposed P/T limits may be incorporated into the St. Lucie 2 TS.

LOW TEMPERATURE OVERPRESSURE PROTECTION

The RCS P/T limits during plant heatup and cooldown are specified in TS Figures 3.4-2 and 3.4-3 for St. Lucie Unit 2. The P/T curves in the current TS are based on an assumed design basis neutron fluence through 6 EFPY.

By letter dated February 7, 1990, as supplemented June 19, 1990, the licensee provided its updated P/T curves in proposed TS Figures 3.4-2 (for heatup and core critical) and 3.4-3 (for cooldown and inservice testing), revised cooldown rates as a function of indicated reactor coolant temperature in proposed Figure 3.4-4, changes in the values of the RCS cold leg temperature at which LTOP should be enabled, and the justification for the changes. The new P/T curves are based on the irradiation damage prediction methods of RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, U.S. Nuclear Regulatory Commission, May 1988, and are applicable for a period up to 15 EFPY.

LTOP for St. Lucie Unit 2 is provided by the power-operated relief valves (PORVs) on the pressurizer and shutdown cooling relief valves (SDCRVs). These valves are set at pressures low enough to prevent violation of the Appendix G heatup and cooldown curves should an RCS pressure transient occur during low temperature operations. The licensee, in its February 7, 1990 submittal, identified the most limiting overpressure transients in determining the PORV setpoints for LTOP. The PORV setpoint limits have been previously set by analysis of the limiting transients for mass addition and energy addition.

The Limiting Conditions for Operation (LCOs) in TS 3.4.9.3, "Overpressure Protection Systems," currently requires that two PORVs or two SDCRVs shall be operable with the setpoints selected for the low temperature mode of operation. The modified TS 3.4.9.3 LCOs a. and c. maintain the same PORV setpoint and revises the values of the applicable temperatures for PORV use for LTOP through reference to a new Table 3.4-4, "minimum cold leg temperature for PORV use for LTOP." The applicability of TS 3.4.9.3 is modified to remove reference to Mode 3 which is outside the proposed LTOP range.

Two design basis mass addition transient were analyzed for representative inadvertent safety injection actuation assumptions. This transient analysis is typically performed to determine the pressure overshoot past the LTOP setpoint such that the Appendix G curves are not exceeded during the transient.

The energy input transient was analyzed assuming a 40°F temperature difference between the steam generator and the RCS. A reactor coolant pump startup in one loop was assumed in order to maximize the heat transfer effect. As was the case for the mass addition transient, the pressure overshoot is calculated such that the Appendix G P/T curves for Unit 2 are not exceeded.

The licensee's analyses were performed using the same methodology as the prior application for 6 EFPY with some changes in the analysis assumptions. For the revised analysis, the LTOP enable temperatures were determined by following the guidance that for LTOP, the enable temperature is the water temperature corresponding to a metal temperature of at least RT_{NDT} + 90°F at the vessel beltline, which was calculated by the licensee to be less than or equal to 247°F during heatup and less than or equal to 230°F during cooldown. The results indicated that changes in the present PORV setpoint of 470 psia and SDCRV setpoint of 350 psia is not required. The new enable temperatures are identified in revised Table 3.4-3, "Low Temperature RCS Overpressure Protection Range."

Table 3.4-4, "Minimum Cold Leg Temperature for PORV Use for LTOP" of LCO 3.4.9.3 is modified to identify the minimum cold leg temperature for PORV use for LTOP as 165°F during heatup and cooldown. This is the temperature identified for transfer of the LTOP function from the SDCRVs to the PORVs which occurs at a temperature above that required for SDCRV alignment to the RCS.

The licensee-proposed changes in TS 3.4.9.3, Tables 3.4-3 and 3.4-4, and the associated Bases sections reflect the above discussed LTOP alignment temperatures and the heatup and cooldown rates identified by the updated Figures 3.4-2, 3.4-3, and 3.4-4 in TS 3.4.9.1. The staff finds that they are based on applicable regulatory guidance in Standard Review Plan (SRP) 5.2.2, Revision 2, are reasonably conservative, and are acceptable.

Other changes identified by the licensee in its amendment request are changes to Definition 1.16, a deletion of the words "for the applicable operating period" in LCO 3.4.1.3 and clarification of the Applicability in TS 3.4.1.4.1 and 3.4.4 to reflect the modified LTOP range. Also, the action statement of

LCO 3.4.9.1 is revised to make reference to the updated Figures 3.4-2, 3.4-3 and 3.4-4. These changes are judged to be administrative in nature and are acceptable.

Technical Finding

Based on the above evaluation, the staff concludes that the licensee's proposed TS and their associated Bases are acceptable to support the updated P/T limits identified in TS 3.4.9.3 applicable for a period up to 15 EFPY.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that this amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

REFERENCES

1. Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, May 1988.

2. NUREG-0800, Standard Review Plan, Section 5.3.2, Pressure-Temperature Limits.

3. Letter from J. H. Goldberg (FPL) to USNRC Document Control Desk, Subject: "Proposed License Amendment P-T Limits and LTOP Analysis," February 7, 1990.

4. A. L. Lowe, Jr. et. al., "Analysis of Capsule W-83, Florida Power and Light Company, St. Lucie Unit No. 2," BAW-1880, September 1985.

Date: August 1, 1990

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