Docket No. 50-335

June 11, 1990

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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 1 - ISSUANCE OF AMENDMENT RE: PRESSURE/TEMPERATURE (P/T) LIMITS AND LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP) ANALYSIS (TAC NO. 75386)

The Commission has issued the enclosed Amendment No. 104 to Facility Operating License No. DPR-67 for the St. Lucie Plant, Unit No. 1. This amendment consists of changes to the Technical Specifications in response to your application dated December 5, 1989.

This amendment incorporates revised P/T limits and the results of a revised LTOP analysis into the Technical Specifications for St. Lucie Unit 1.

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. 104 to DPR-67
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	LA:PD22	PM:PD22	D:PD22	OGC			
NAME	DMiller	JNorris	HBerkow	[illegible]			
DATE	5/29/90	5/29/90	5/30/90	6/1/90			

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Mr. J. H. Goldberg
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. 104
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 December 5, 1989, 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. 104 , 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: June 11, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 104

TO FACILITY OPERATING LICENSE NO. DPR-67

DOCKET NO. 50-335

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 1-8	3/4 1-8
3/4 1-9a	3/4 1-9a
3/4 1-12	3/4 1-12
3/4 4-21	3/4 4-21
3/4 4-23a	3/4 4-23a
3/4 4-23b	3/4 4-23b
3/4 4-23c	3/4 4-23c
3/4 4-59	3/4 4-59
3/4 4-60	3/4 4-60
3/4 5-7	3/4 5-7
B 3/4 4-7	B 3/4 4-7
B 3/4 4-15	B 3/4 4-15
B 3/4 5-1	B 3/4 5-1

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/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 RCS OVERPRESSURE PROTECTION RANGE

1.16 The LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE is that operating condition when (1) the cold leg temperature is < 304°F during heatup or ≤ 281°F during cooldown and (2) the Reactor Coolant System has pressure boundary integrity. The Reactor Coolant System does not have pressure boundary integrity when the Reactor Coolant System is open to containment and the minimum area of the Reactor Coolant System opening is greater than 1.75 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 calculations 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 Sample point locations.

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.5 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be $\geq$ 515°F when the reactor is critical.

APPLICABILITY: MODES 1 and 2#.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) < 515°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.5 The Reactor Coolant System temperature (T_{avg}) shall be determined to be $\geq$ 515°F.

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System temperature (T_{avg}) is < 525°F.

With $K_{eff} \geq 1.0$.

3/4.1.2 BORATION SYSTEMS

FLOW PATHS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

a. A flow path from the boric acid makeup tank via either a boric acid pump or a gravity feed connection and any charging pump to the Reactor Coolant System if only the boric acid makeup tank in Specification 3.1.2.7a is OPERABLE, or

b. The flow path from the refueling water tank via either a charging pump or a high pressure safety injection pump* to the Reactor Coolant System if only the refueling water tank in Specification 3.1.2.7b is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one injection path is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

*The flow path from the RWT to the RCS via a single HPSI pump shall only be established if: (a) the RCS pressure boundary does not exist, or (b) no charging pumps are operable. In this case all charging pumps shall be disabled, and heatup and cooldown rates shall be limited in accordance with Fig. 3.1-1b.

At RCS temperatures below 115°F, any two of the following valves in the operable HPSI header shall be verified closed and have their power removed:

High Pressure Header	Auxiliary Header
HCV-3616	HCV-3617
HCV-3626	HCV-3627
HCV-3636	HCV-3637
HCV-3646	HCV-3647

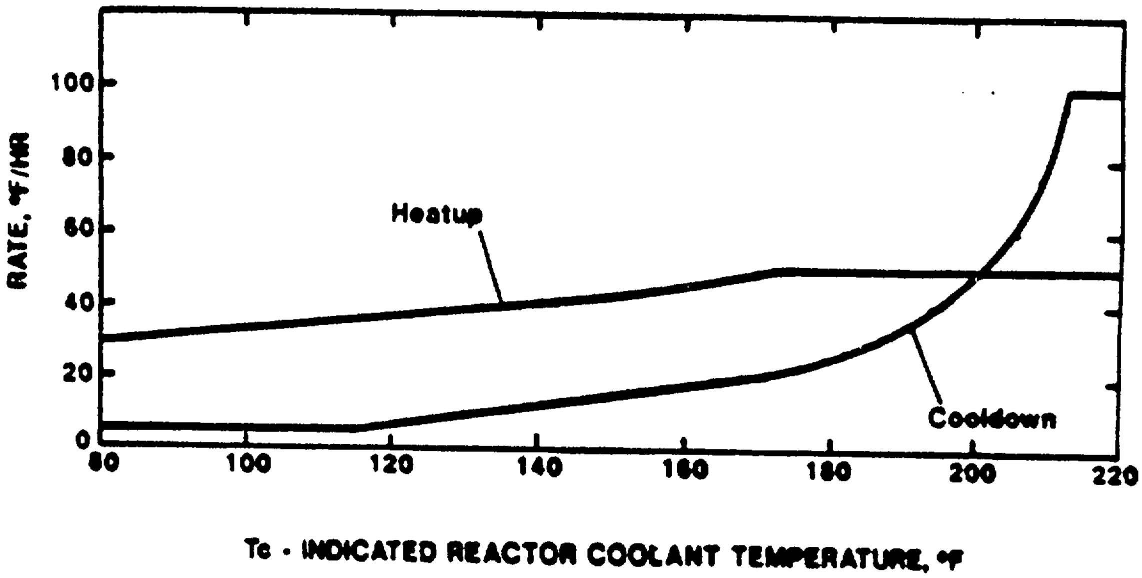

FIGURE 3.1-1b
MAXIMUM ALLOWABLE HEATUP AND COOLDOWN RATES,
SINGLE HPSI PUMP IN OPERATION

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CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump or one high pressure safety injection pump* in the boron injection flow path required OPERABLE pursuant to Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump or high pressure safety injection pump*OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one of the required pumps is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.3 At least one of the above required pumps shall be demonstrated OPERABLE by verifying the charging pump develops a flow rate of greater than or equal to 40 gpm or the high pressure safety injection pump develops a total head of greater than or equal to 2571 ft. when tested pursuant to Specification 4.0.5.

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*The flow path from the RWT to the RCS via a single HPSI pump shall be established only if: (a) the RCS pressure boundary does not exist, or (b) no charging pumps are operable. In this case, all charging pumps shall be disabled and heatup and cooldown rates shall be limited in accordance with Fig. 3.1-1b.

At RCS temperatures below 115°F, any two of the following valves in the operable HPSI header shall be verified closed and have their power removed:

High Pressure Header	Auxiliary Header
HCV-3616	HCV-3617
HCV-3626	HCV-3627
HCV-3636	HCV-3637
HCV-3646	HCV-3647

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 3.4-2a, 3.4-2b and 3.4-3 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 limits within 30 minutes; perform an analysis to determine the effects of the out-of-limit condition on the fracture toughness properties 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 following 30 hours in accordance with Figures 3.4-2b and 3.4-3.

*When the flow path from the RWT to the RCS via a single HPSI pump is established per 3.1.2.3, the heatup and cooldown rates shall be established in accordance with Fig. 3.1-1b.

#During hydrostatic testing operations above system design pressure, a maximum temperature change in any one hour period shall be limited to 5°F.

SURVEILLANCE REQUIREMENTS

4.4.9.1

a. 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.

b. The Reactor Coolant System temperature and pressure conditions shall be determined to be to the right of the criticality limit line within 15 minutes prior to achieving reactor criticality.

c. The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals shown in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2a, 3.4-2b and 3.4-3.

FIGURE 3.4-2a
ST. LUCIE UNIT 1 P/T LIMITS, 15 EFPY
HEATUP AND CORE CRITICAL

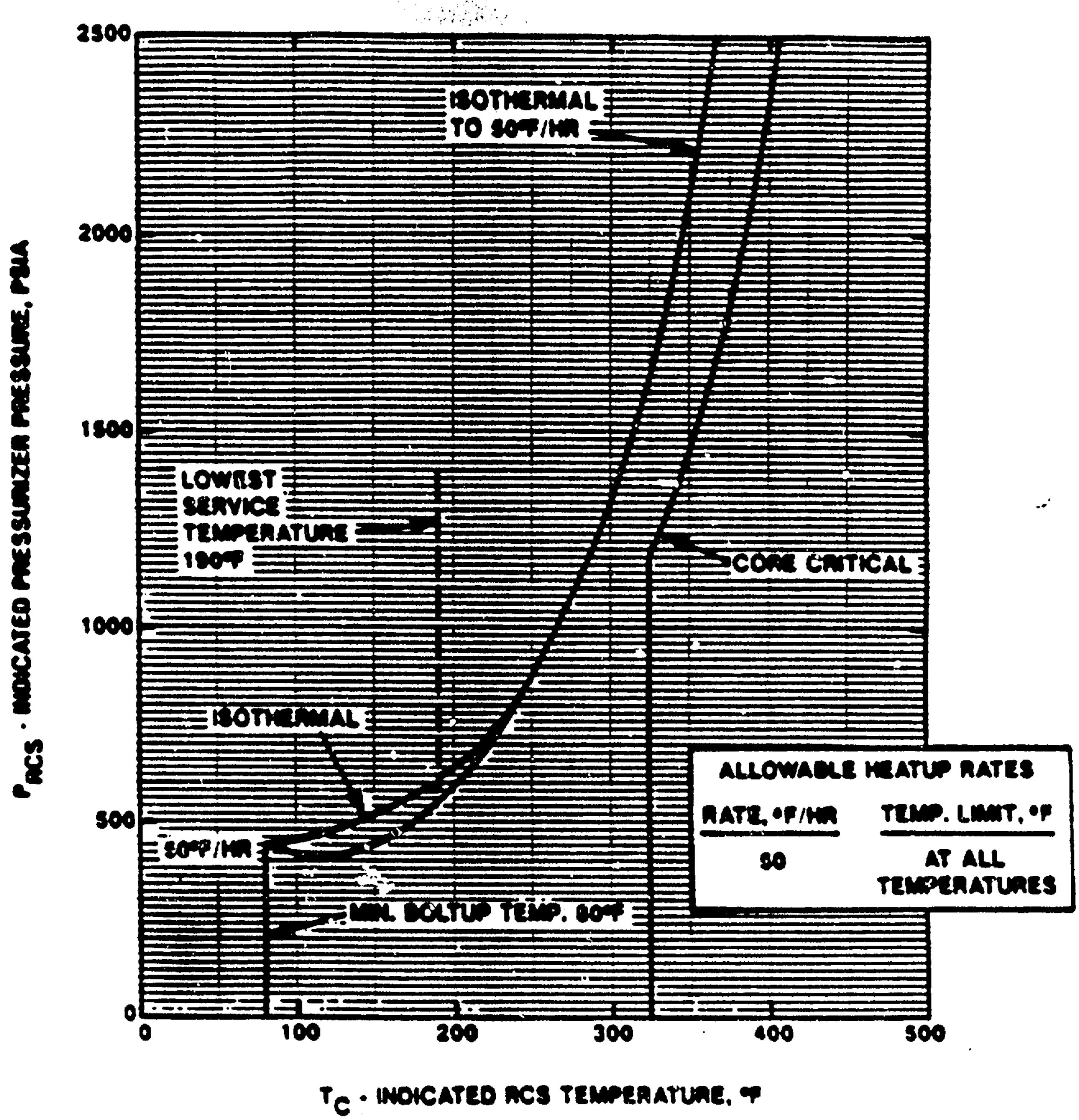


FIGURE 3.4-2b

ST. LUCIE UNIT 1 P/T LIMITS, 15 EFPY
COOLDOWN AND INSERVICE TEST

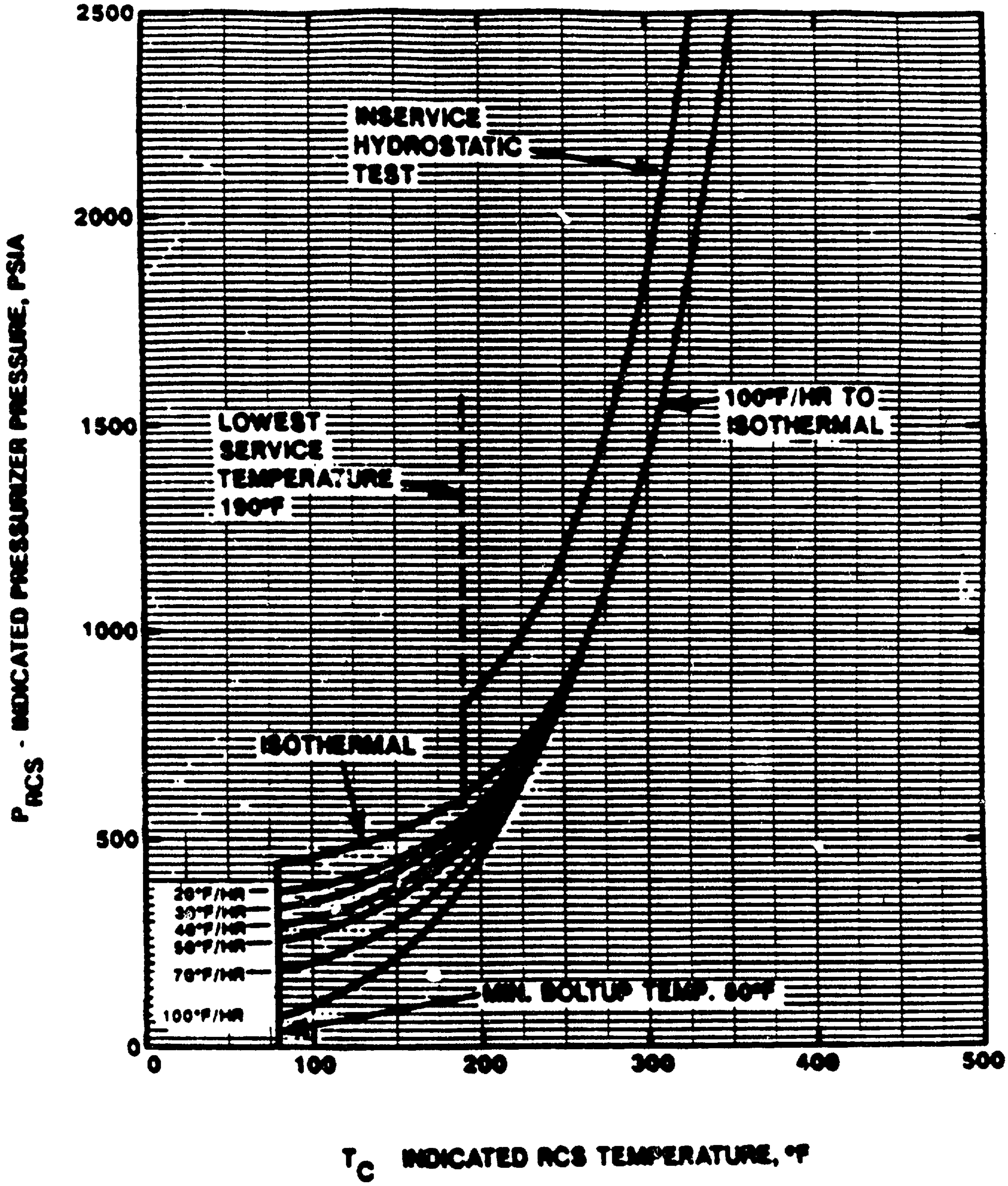
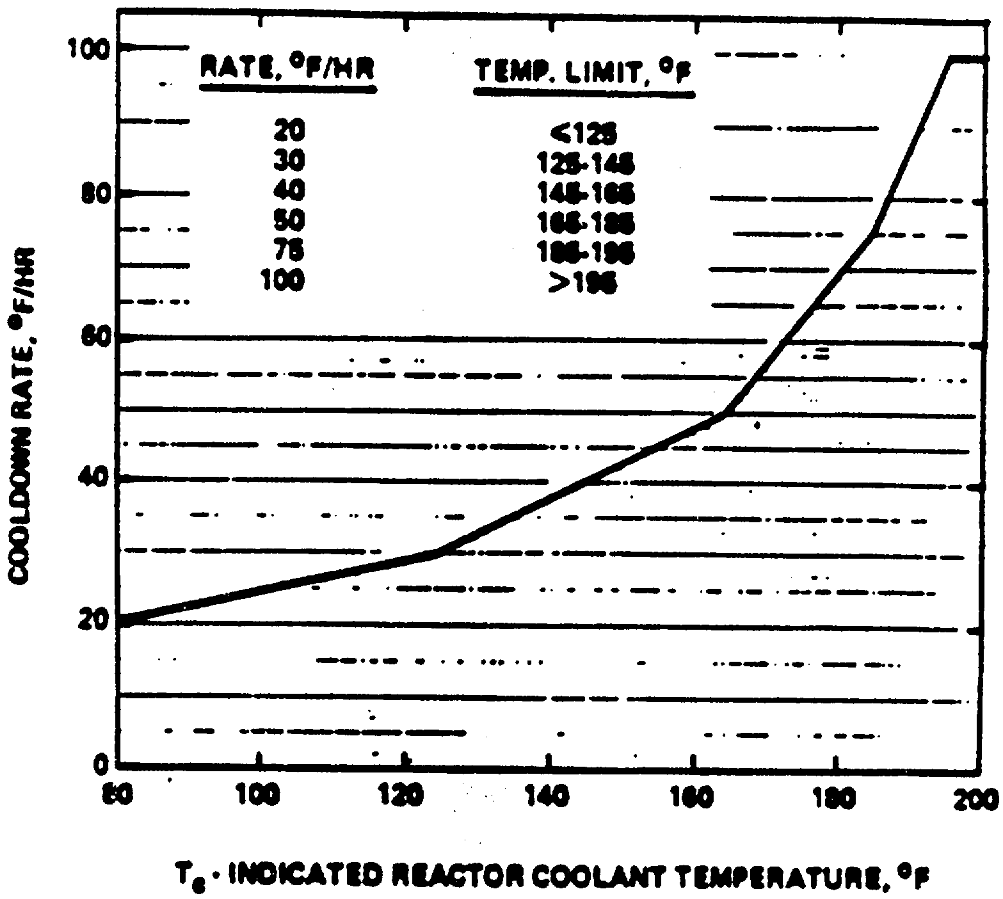


FIGURE 3.4-3
ST. LUCIE UNIT 1, 15 EFPY
MAXIMUM ALLOWABLE COOLDOWN RATES



NOTE: A MAXIMUM COOLDOWN RATE OF 100°F/HR IS ALLOWED AT ANY TEMPERATURE ABOVE 195°F

POWER OPERATED RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.13 Two power operated relief valves (PORVs) shall be OPERABLE, with their setpoints selected to the low temperature mode of operation as follows:

a. A setpoint of less than or equal to 350 psia shall be selected:

1. During cooldown when the temperature of any RCS cold leg is less than or equal to 215°F and

2. During heatup and isothermal conditions when the temperature of any RCS cold leg is less than or equal to 193°F.

b. A setpoint of less than or equal to 530 psia shall be selected:

1. During cooldown when the temperature of any RCS cold leg is greater than 215°F and less than or equal to 281°F.

2. During heatup and isothermal conditions when the temperature of any RCS cold leg is greater than or equal to 193°F and less than or equal to 304°F.

APPLICABILITY: MODES 4# and 5*.

ACTION:

a. With less than two PORVs OPERABLE and while at Hot Shutdown during a planned cooldown, both PORVs will be returned to OPERABLE status prior to entering the applicable MODE unless:

1. The repairs cannot be accomplished within 24 hours or the repairs cannot be performed under hot conditions, or

2. Another action statement requires cooldown, or

3. Plant and personnel safety requires cooldown to Cold Shutdown with extreme caution.

b. With less than two PORVs OPERABLE while in COLD SHUTDOWN, both PORVs will be returned to OPERABLE status prior to startup.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.13 The PORVs shall be verified OPERABLE by:

a. Verifying the isolation valves are open when the PORVs are reset to the low temperature mode of operation.

b. Performance of a CHANNEL FUNCTIONAL TEST of the Reactor Coolant System overpressurization protection system circuitry up to and including the relief valve solenoids once per refueling outage.

c. Performance of a CHANNEL CALIBRATION of the pressurizer pressure sensing channels once per 18 months.

#Reactor Coolant System cold leg temperature below 304°F.

*PORVs are not required below 140°F when RCS does not have pressure boundary integrity.

REACTOR COOLANT SYSTEM

REACTOR COOLANT PUMP - STARTING

LIMITING CONDITION FOR OPERATION

3.4.14 If the steam generator temperature exceeds the primary temperature by more than 30°F, the first idle reactor coolant pump shall not be started.

APPLICABILITY: MODES 4# and 5.

ACTION:

If a reactor coolant pump is started when the steam generator temperature exceeds primary temperature by more than 30°F, evaluate the subsequent transient to determine compliance with Specification 3.4.9.1.

SURVEILLANCE REQUIREMENTS

4.4.14 Prior to starting a reactor coolant pump, verify that the steam generator temperature does not exceed primary temperature by more than 30°F.

#Reactor Coolant System Cold Leg Temperature is less than 304°F.

ECCS SUBSYSTEMS - T_{avg} < 325°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. In MODES 3* and 4#, one ECCS subsystem composed of one OPERABLE high pressure safety injection pump and one OPERABLE flow path capable of taking suction from the refueling water storage tank on a safety injection actuation signal and automatically transferring suction to the containment sump on a sump recirculation actuation signal.

b. Prior to decreasing the reactor coolant system temperature below 270°F a maximum of only one high pressure safety injection pump shall be OPERABLE with its associated header stop valve open.

c. Prior to decreasing the reactor coolant system temperature below 236°F all high pressure safety injection pumps shall be disabled and their associated header stop valves closed except as allowed by Specifications 3.1.2.1 and 3.1.2.3.

APPLICABILITY: MODES 3* and 4#.

ACTION:

a. With no ECCS subsystems OPERABLE in MODES 3* and 4#, immediately restore one ECCS subsystem to OPERABLE status or be in COLD SHUTDOWN within 20 hours.

b. With RCS temperature below 270°F and with more than the allowed high pressure safety injection pump OPERABLE or injection valves and header isolation valves open, immediately disable the high pressure safety injection pump(s) or close the header isolation valves.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

4.5.3.2 The high pressure safety injection pumps shall be verified inoperable and the associated header stop valves closed prior to decreasing below the above specified Reactor Coolant System temperature and once per month when the Reactor Coolant System is at refueling temperatures.

*With pressurizer pressure < 1750 psia.
#REACTOR COOLANT SYSTEM cold leg temperature above 250°F.

REFUELING WATER TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water tank shall be OPERABLE with:

a. A minimum contained volume 401,800 gallons of borated water,

b. A minimum boron concentration of 1720 ppm,

c. A maximum water temperature of 100°F,

d. A minimum water temperature of 55°F when in MODES 1 and 2, and

e. A minimum water temperature of 40°F when in MODES 3 and 4

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWT shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the water level in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWT temperature.

The heatup and cooldown limit curves (Figures 3.4-2a and 3.4-2b) 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°F/hr and for any cooldown rate of up to 100°F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the applicable service period.

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>1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature can be calculated based upon the fluence. The heatup and cooldown limit curves shown on Figures 3.4-2a and 3.4-2b include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for pressure differences between the reactor vessel beltline and pressurizer instrument taps.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-82, reactor vessel material surveillance specimens installed near the inside wall of the reactor vessel in the core area. The capsules are scheduled for removal at times that correspond to key accumulated fluence levels within the vessel through the end of life. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, measured ΔRT_{NDT} for surveillance samples can be applied with confidence to the corresponding material in the reactor vessel wall. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figures 3.4-2a and 3.4-2b for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements for 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 estimated to be 90°F. The Lowest Service Temperature limit line shown on Figures 3.4-2a and 3.4-2b is based upon this RT_{NDT} since Article NB-2332 of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

COMPONENT	COMP CODE	MATERIAL TYPE	CU %	NI %	P %	NDTT F	50 FT-LB/35 MIL TEMP F LONG(1)	50 FT-LB/35 MIL TEMP F TRANS(1,2)	RTNDT(4) F	MIN. UPPER SHELF FT-LB LONG	MIN. UPPER SHELF FT-LB TRANS(3)
Vessel Flange Forging	C-1-1	A508Cl.2	-	-	.008	+20	+70	+90	+30	133	86
Bottom Head Plate	C-10-1	A533BCl.1	-	-	.010	-40	+42	+62	+2	120	78
Bottom Head Plate	C-9-2	A533BCl.1	-	-	.011	-40	-18	+2	-40	146	95
Bottom Head Plate	C-9-3	A533BCl.1	-	-	.013	-70	-20	0	-60	148	96
Bottom Head Plate	C-9-1	A533BCl.1	-	-	.011	-30	+10	+30	-30	138	90
Inlet Nozzle	C-4-3	A508Cl.2	-	-	.005	0	0	+20	0	111	72
Inlet Nozzle	C-4-2	A508Cl.2	-	-	.004	0	+20	+40	0	146	95
Inlet Nozzle	C-4-1	A508Cl.2	-	-	.005	+10	-25	-5	10	144	94
Inlet Nozzle	C-4-4	A508Cl.2	-	-	.004	0	+10	+30	0	139	90
Inlet Nozzle Ext.	C-16-3	A508Cl.2	-	-	.001	+10	+52	+72	+12	139	90
Inlet Nozzle Ext.	C-16-2	A508Cl.2	-	-	.011	+10	+52	+72	+12	139	90
Inlet Nozzle Ext.	C-16-1	A508Cl.2	-	-	.011	+10	+52	+72	+12	139	90
Inlet Nozzle Ext.	C-16-4	A508Cl.2	-	-	.011	+10	+52	+72	+12	139	90

3/4.4.13 POWER OPERATED RELIEF VALVES and 3/4.4.14 REACTOR COOLANT PUMP - STARTING

The low temperature overpressure protection system (LTOP) is designed to prevent RCS overpressurization above the 10 CFR Appendix G operating limit curves (Figures 3.4-2a and 3.4-2b) at RCS temperatures at or below 304°F during heatup and 281°F during cooldown. The LTOP system is based on the use of the pressurizer power-operated relief valves (PORVs) and the implementation of administrative and operational controls.

The PORVs aligned to the RCS with the low pressure setpoints of 350 and 530 psia, restrictions on RCP starts, limitations on heatup and cooldown rates, and disabling of non-essential components provide assurance that Appendix G P/T limits will not be exceeded during normal operation or design basis overpressurization events due to mass or energy addition to the RCS.

3/4.4.15 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 irreverisible 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.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

BASES

3/4.5.1 SAFETY INJECTION TANKS

The OPERABILITY of each of the RCS safety injection tanks ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the safety injection tanks. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on safety injection tank volume, boron concentration and pressure ensure that the assumptions used for safety injection tank injection in the accident analysis are met.

The limit of one hour for operation with an inoperable safety injection tank minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional safety injection tank which may result in unacceptable peak cladding temperatures.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate and independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the safety injection tanks is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the accident analyses are met and that subsystem OPERABILITY is maintained.

The limitations on HPSI pump operability when the RCS temperature is $\leq$ 270°F and $\leq$ 236°F, and the associated Surveillance Requirements provide additional administrative assurance that the pressure/temperature limits (Figures 3.4-2a and 3.4-2b) will not be exceeded during a mass addition transient mitigated by a single PORV.

3/4.5.4 REFUELING WATER TANK (RWT)

The OPERABILITY of the RWT as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWT minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWT and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.104

TO FACILITY OPERATING LICENSE NO. DPR-67

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT, UNIT NO. 1

DOCKET NO. 50-335

INTRODUCTION

By application dated December 5, 1989, 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 1. The current St. Lucie Unit 1 TS for P/T and LTOP are applicable to 10 effective full power years (EFPY). Accordingly, the St. Lucie 1 TS require revision prior to the plant reaching 10 EFPY. Below is the staff's evaluation of the proposed changes.

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 1 TS, Section 3.4. This revision also changes the effectiveness of the P/T limits from 10 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 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 1 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 for St. Lucie 1 at 1/4T (T = vessel beltline thickness) was the lower shell longitudinal weld seams (3-203A, B, and C) with 0.30% copper (Cu), 0.64% nickel (Ni), and an initial RT_{ndt} of -56°F. At 3/4T, the material with the highest ART at 15 EFPY was lower shell plate C-5935-1 with 0.15% Cu, 0.57% Ni, and an initial RT_{ndt} of 20°F.

The licensee has removed one surveillance capsule from St. Lucie 1. The results from capsule W-97 were published in Combustion Engineering Report TR-F-MCM-004. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material at 1/4T, the lower shell longitudinal weld seams, the staff calculated the ART to be 181.4°F at 15 EFPY. For the limiting beltline material at 3/4T, lower shell plate C-5935-1, the staff calculated the ART to be 137.2°F. The staff used a neutron fluence of 1.22E19 n/cm² at 1/4T and 4.34E18 n/cm² at 3/4T. The ART was determined using Section 1 of RG 1.99, Rev. 2 because only one surveillance capsule has been removed from the St. Lucie 1 reactor vessel.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 191°F at 1/4T and 137°F at 3/4T. The licensee's ART of 191°F is more conservative than the staff's ART of 181.4°F; therefore it is acceptable. Substituting the ART of 191°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 30°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 USE data for all plate materials and one weld in the beltline area, but does not have any USE data for the intermediate shell longitudinal weld seams (2-203A, B, and C) and the lower shell longitudinal weld seams (3-203A, B, and C). The staff will obtain the USE of these two welds in the near future. Presently, the staff has determined that all the materials for which USE data are available will meet the requirement that the Charpy USE at end of life 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 1 TS.

LOW TEMPERATURE OVERPRESSURE PROTECTION

The Reactor Coolant System (RCS) P/T limits during plant heatup and cooldown are specified in Technical Specification Figures 3.4-2a and 3.4-2b for St. Lucie Unit 1. The P/T curves in the current TS are based on an assumed design basis neutron fluence through 10 EFPY.

By letter dated December 5, 1989, the licensee provided its updated P/T curves in proposed TS Figures 3.4-2a (for heatup and core critical) and 3.4-2b (for cooldown and inservice testing), 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. New cooldown rates as a function of indicated reactor coolant temperature are also proposed in an updated Figure 3.4-3.

LTOP is provided by the PORVs on the pressurizer. These PORVs 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 December 5, 1989 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. TS 3.4.13 maintains the same pressure setpoints and revises the values of the applicable temperatures for LTOP. A modification to TS 3.4.14, "Reactor Coolant Pump - Starting," identifies the new applicable temperature for LTOP in Mode 4.

The most limiting mass addition transient was analyzed assuming a spectrum of inadvertent safety injection actuation assumptions. The 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 30°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 1 are not exceeded.

The present TS 3.5.3, "ECCS Subsystems - T_{avg} 325°F," allows only one high pressure safety injection (HPSI) pump to be operable prior to decreasing the RCS temperature below 253°F and requires disabling of all HPSI pumps prior to decreasing the RCS temperature below 220°F. In the proposed revision, the RCS temperature in LCO 3.5.3b and Action b would change from 253°F to 270°F, and the RCS temperature in LCO 3.5.3c would change from 220°F to 236°F. These changes are necessary as a result of a reanalysis of inadvertent safety injection actuation in a LTOP condition. During heatup, a HPSI pump may be returned to service at 236°F. To provide a reasonable operational margin for returning a HPSI pump to service, the applicability of 3.5.3 in Mode 4 would be revised from a RCS cold leg temperature above 235°F to a RCS cold leg temperature above 250°F. This change is made to a footnote in TS 3.5.3.

The licensee's analyses were performed using the same methodology as the prior application for 10 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 304°F during heatup and 281°F during cooldown. The results indicated that a change in the present PORV setpoints of 350 psia and 530 psia is not required.

The Definitions Section 1.16, "Low Temperature RCS Overpressure Protection Range," is revised to identify the cold leg temperature for the LTOP range as less than or equal to 304°F during heatup or less than or equal to 281°F during cooldown.

The licensee-proposed changes in TS 1.16, 3.4.13, 3.4.14, and 3.5.3, and the associated Bases sections reflect the above discussed LTOP alignment temperatures and the heatup and cooldown rates identified by the updated Figures 3.4-2a, 3.4-2b, and 3.4-3 in TS 3.4.9. The staff finds that they are reasonably conservative and acceptable.

Technical Finding

Based on the above evaluation, the staff concludes that the proposed TS 1.16, 3.4.13, 3.4.14, 3.5.3, and their associated Bases are acceptable to support the updated P/T limits identified in TS 3.4.9.1 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: "St. Lucie 1 Proposed License Amendment P-T Limits and LTOP Analysis," December 9, 1989.

4. S. T. Byrne, "Florida Power and Light Company St. Lucie Unit No. 1, Post-Irradiation Evaluation of Reactor Vessel Surveillance Capsule W-97," Combustion Engineering Report No. TR-F-MCM-004, December 1983.

Date: June 11, 1990

Principal Contributors:
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DATED: June 11, 1990

AMENDMENT NO. 104 TO FACILITY OPERATING LICENSE NO. DPR-67 - ST. LUCIE, UNIT 1

Docket File
NRC & Local PDRs
PDII-2 Reading
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G. Lainas, 14/H/3
H. Berkow
D. Miller
J. Norris
OGC-WF
D. Hagan, 3302 MNBB
E. Jordan, 3302 MNBB
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Others as required

cc: Plant Service list