June 5, 1987

Docket No. 50-335

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Mr. C. O. Woody
Group Vice President
Nuclear Energy
Florida Power & Light Company
P. O. Box 14000
Juno Beach, Florida 33408

Dear Mr. Woody:

The Commission has issued the enclosed Amendment No. 81 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 March 17, 1987.

This amendment changes the Reactor Coolant System Pressure/Temperature (P/T) limit figures to be effective up to ten (10) effective full power years of operation. The amendment also changes the technical specifications dealing with overpressure protection systems because they are linked with the new P/T limit figures. The applicable bases sections are changed to reflect the above changes.

A copy of the related Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's next bi-weekly Federal Register notice.

Sincerely,

/s/

E. G. Tourigny, Project Manager
Project Directorate II-2
Division of Reactor Projects-I/II

Enclosures:
1. Amendment No. 81 to DPR-67
2. Safety Evaluation

cc w/enclosures:
See next page

LA:PD22	PM:PD22	BC:RSB	OGC	D:PD22
DMiller	ETourigny:hc	WHodges	w/correction to SE Woodhead	LRubenstein
6/2/87	6/2/87	6/3/87	6/3/87	6/5/87

Mr. C. O. Woody
Florida Power & Light Company

St. Lucie Plant

cc:
Mr. Jack Shreve
Office of the Public Counsel
Room 4, Holland Building
Tallahassee, Florida 32304

Resident Inspector
c/o U.S. NRC
7585 S. Hwy A1A
Jensen Beach, Florida 33457

State Planning & Development
Clearinghouse
Office of Planning & Budget
Executive Office of the Governor
The Capitol Building
Tallahassee, Florida 32301

Harold F. Reis, Esq.
Newman & Holtzinger
1615 L Street, N.W.
Washington, DC 20036

Norman A. Coll, Esq.
McCarthy, Steel, Hector and Davis
14th Floor, First National Bank Building
Miami, Florida 33131

Administrator
Department of Environmental Regulation
Power Plant Siting Section
State of Florida
2600 Blair Stone Road
Tallahassee, Florida 32301

Mr. Weldon B. Lewis, County
Administrator
St. Lucie County
2300 Virginia Avenue, Room 104
Fort Pierce, Florida 33450

Mr. Charles B. Brinkman, Manager
Washington - Nuclear Operations
Combustion Engineering, Inc.
7910 Woodmont Avenue
Bethesda, Maryland 20814

Jacob Daniel Nash
Office of Radiation Control
Department of Health and
Rehabilitative Services
1317 Winewood Blvd.
Tallahassee, Florida 32399-0700

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
Executive Director for Operations
101 Marietta Street N.W., Suite 2900
Atlanta, Georgia 30323

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. 81
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 March 17, 1987, 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. 81 , 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

Lester S. Rubenstein, Director
Project Directorate II-2
Division of Reactor Projects-I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 5, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 81

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 1-9b
3/4 1-12	3/4 1-12
3/4 4-1d	3/4 4-1d
3/4 4-21	3/4 4-21
3/4 4-22	3/4 4-22
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
B3/4 1-3	B3/4 1-3
B3/4 4-1	B3/4 4-1
B3/4 4-6	B3/4 4-6
B3/4 4-7	B3/4 4-7
B3/4 4-8	B3/4 4-8
B3/4 4-9	B3/4 4-9
B3/4 4-10	B3/4 4-10
B3/4 4-11	B3/4 4-11
B3/4 4-15	B3/4 4-15
B3/4 5-1	B3/4 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 ≤ 334°F 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 and one associated heat tracing circuit shall be OPERABLE:

a. A flow path from the boric acid makeup tank via either a boric acid pump or a gravity feed connection and 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 7 days by:

1. Cycling each testable power operated or automatic valve in the flow path required for boron injection through at least one complete cycle of full travel, and

2. Verifying that the temperature of the heat traced portion of the flow path is above the temperature limit line shown on Figure 3.1-1 when a flow path from the boric acid make-up tanks is used.

*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.

SURVEILLANCE REQUIREMENTS

b. 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.

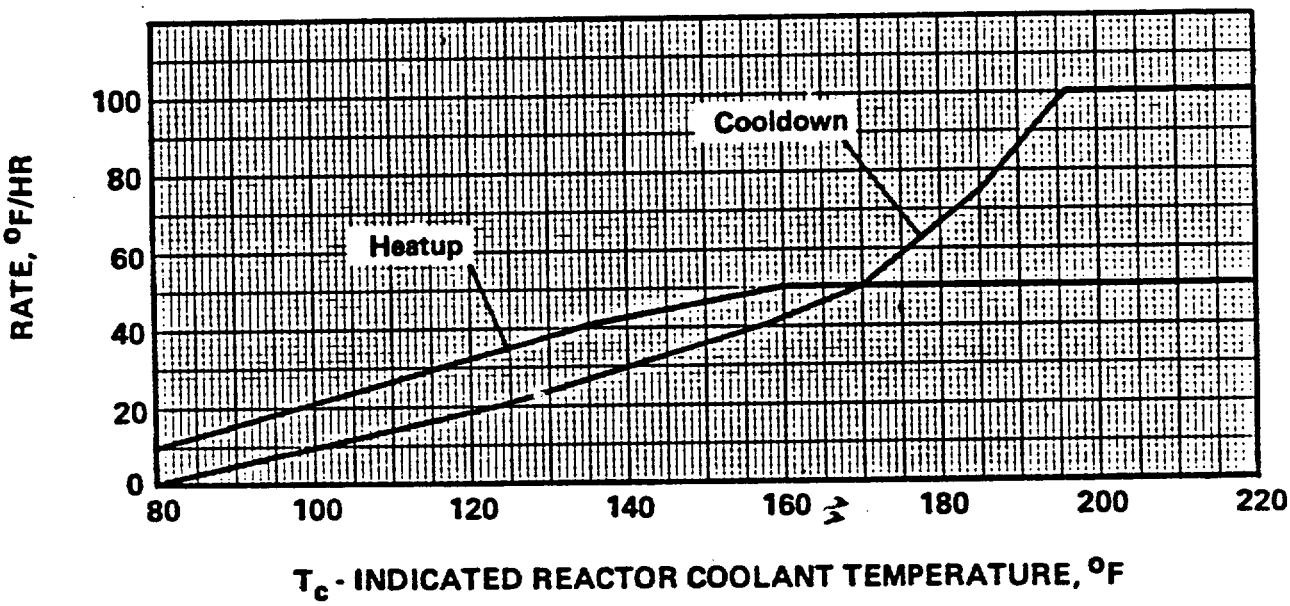

FIGURE 3.1-1b
MAXIMUM ALLOWABLE HEATUP AND COOLDOWN RATES, SINGLE HPSI PUMP IN OPERATION

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FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths and one associated heat tracing circuit shall be OPERABLE:

a. Two flow paths from the boric acid makeup tanks via either a boric acid pump or a gravity feed connection, and a charging pump to the Reactor Coolant System, and

b. The flow path from the refueling water tank via a charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or make the reactor subcritical within the next 2 hours and borate to a SHUTDOWN MARGIN equivalent to at least 2% $\Delta k/k$ at 200°F; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Cycling each testable power operated or automatic valve in the flow path through at least one complete cycle of full travel.

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 the above required charging pump or high pressure safety injection pump shall be demonstrated OPERABLE at least once per 31 days by:

a. Starting (unless already operating) the pump from the control room,

b. Verifying pump operation for at least 15 minutes, and

c. Verifying that the pump is aligned to receive electrical power from an OPERABLE emergency bus.

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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.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the temperature of the heat traced portion of the flow path from the boric acid makeup tanks is above the temperature limit line shown on Figure 3.1-1.

b. 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.

c. At least once per 18 months during shutdown by:

1. Cycling each power operated (excluding automatic) valve in the flow path that is not testable during plant operation, through at least once complete cycle of full travel.

2. Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection Actuation signal.

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% of narrow range indication.

APPLICABILITY: MODE 5 with reactor coolant loops filled##.

ACTION:

a. With less than the above required loops OPERABLE or with less than the required steam generator level, within one (1) hour initiate corrective action to return the required loops to OPERABLE status or to restore the required level.

b. With no shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and within one (1) hour 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 Normal or Emergency Power Source may be inoperable for each shutdown cooling loop.

**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 30°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 one (1) hour initiate corrective action to return the required loops to OPERABLE status.

b. With no shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and within one (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 Normal or Emergency Power Source may be inoperable for each shutdown cooling loop.

*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.

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} and pressure to less than 200°F and 500 psia, respectively, within the following 30 hours.

*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, 10 EFPY
HEATUP AND CORE CRITICAL

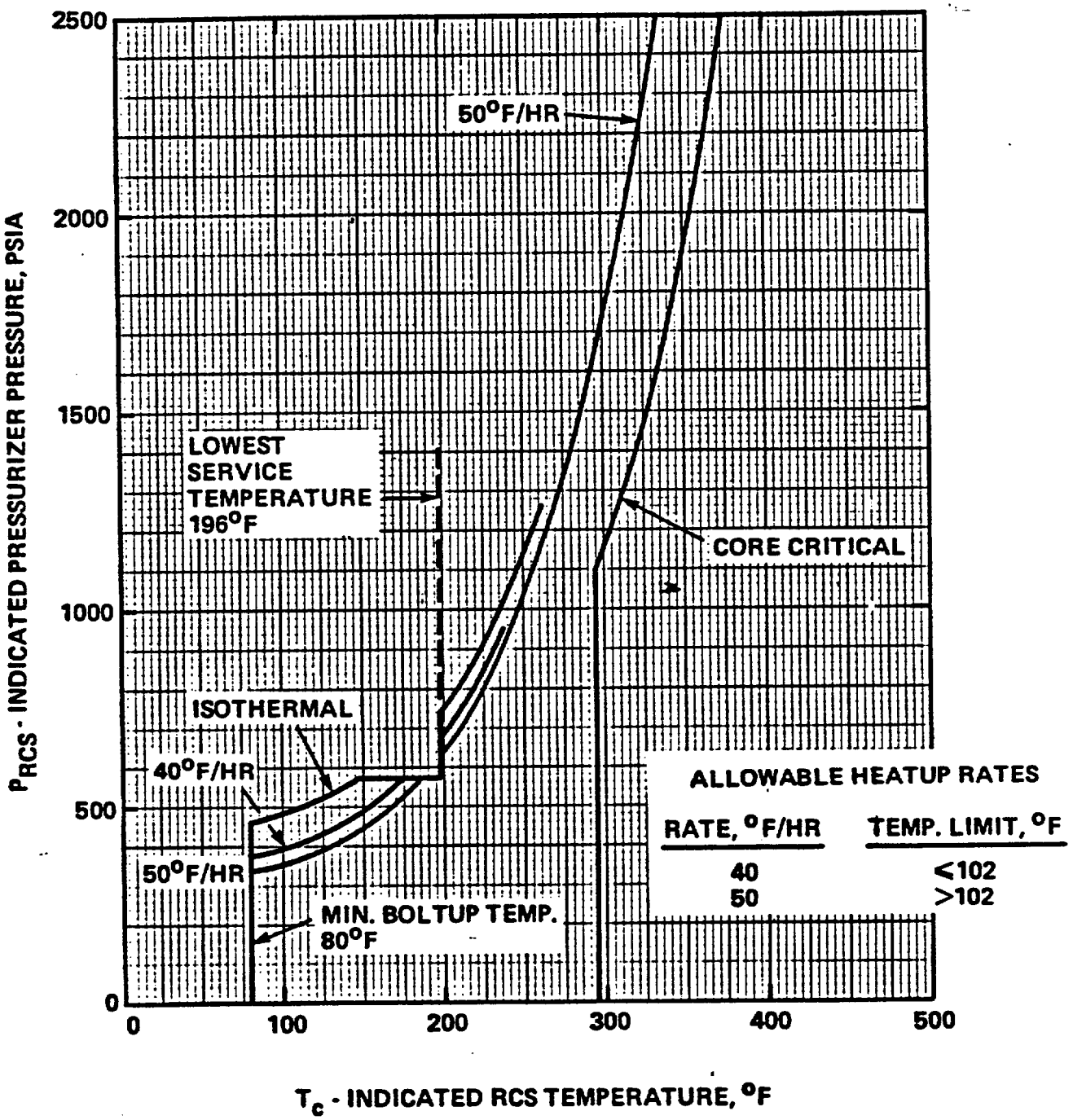


FIGURE 3.4-2b
ST. LUCIE UNIT 1 P/T LIMITS, 10 EFPY
COOLDOWN AND INSERVICE TEST

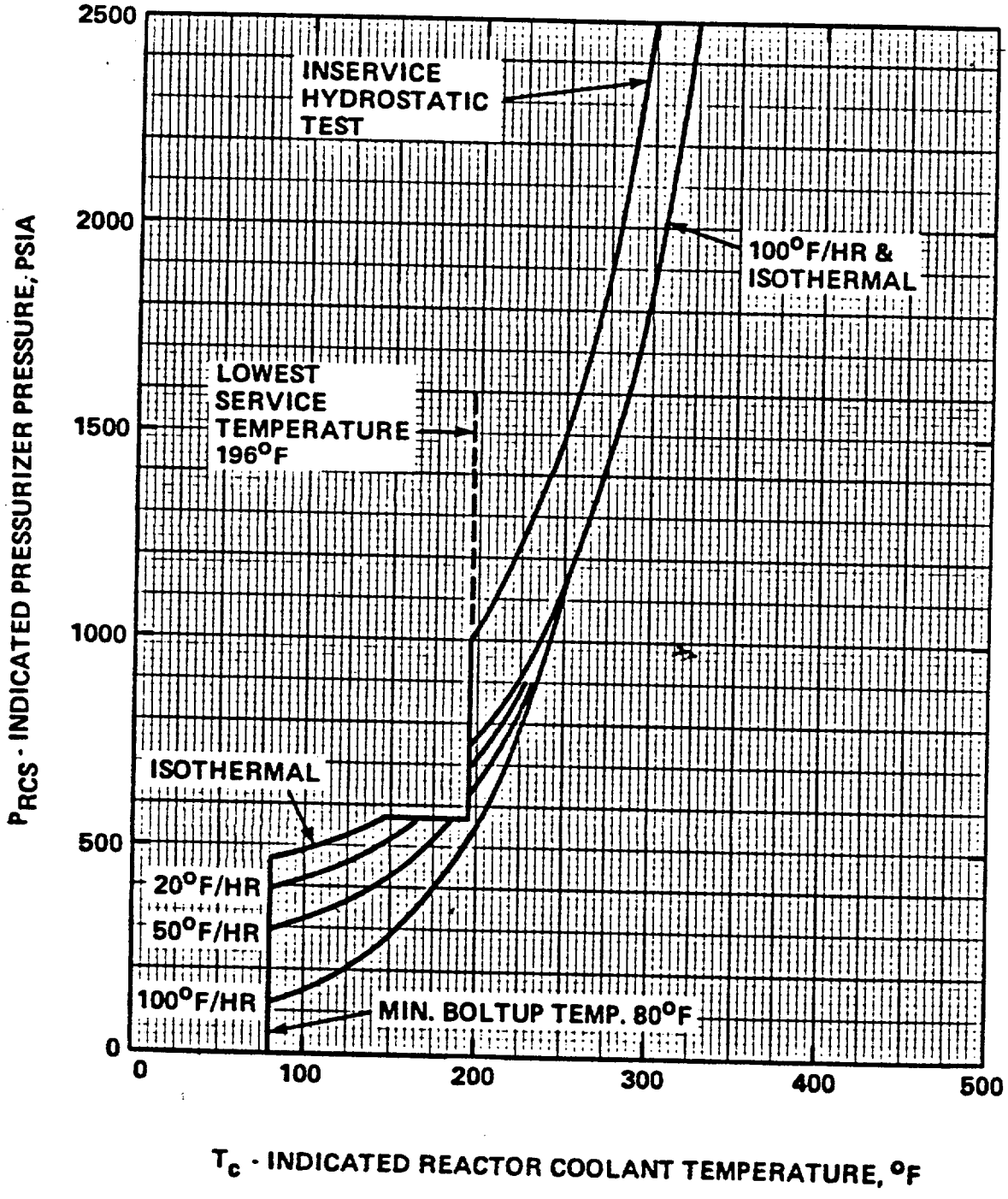
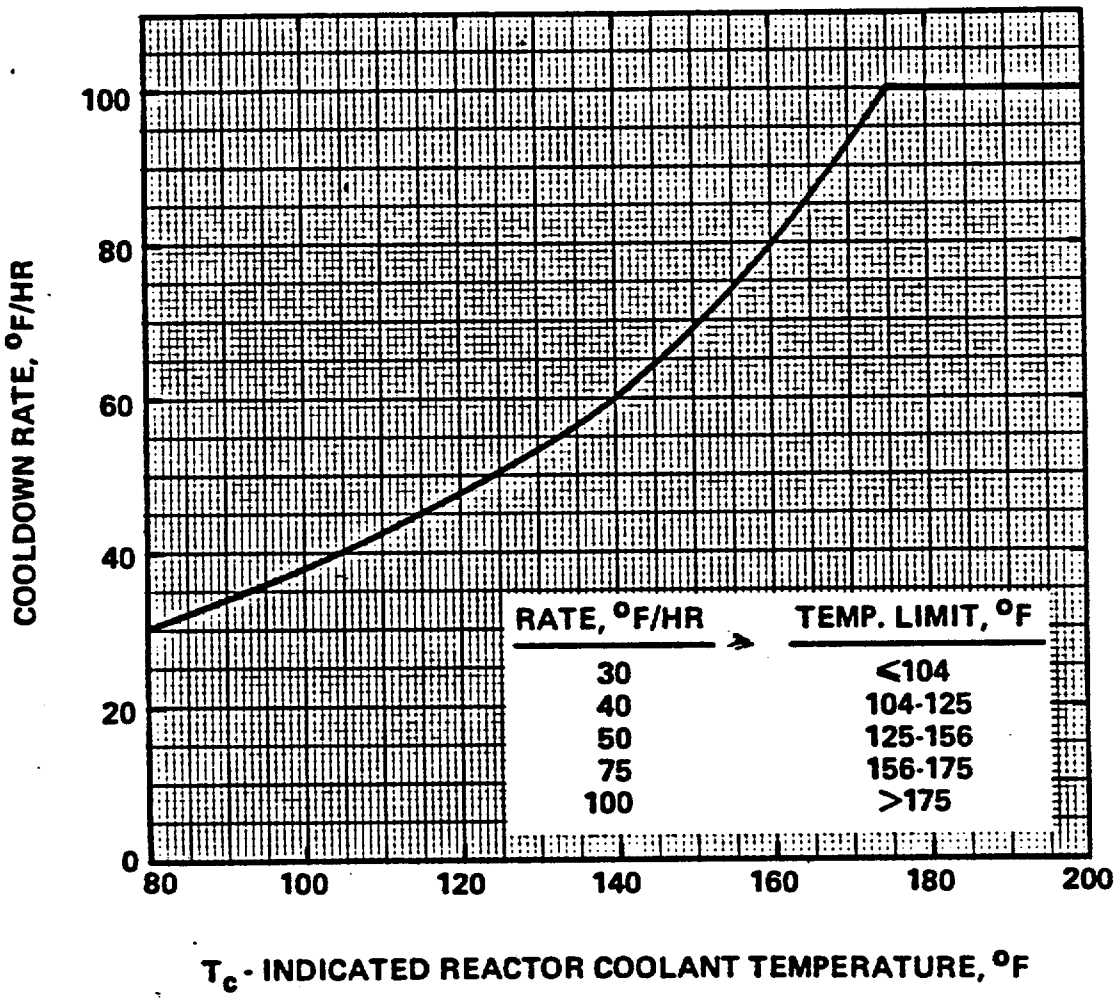


FIGURE 3.4-3
ST. LUCIE UNIT 1, 10 EFPY
MAXIMUM ALLOWABLE COOLDOWN RATES



NOTE: A MAXIMUM COOLDOWN RATE OF 100°F/HR IS ALLOWED AT ANY TEMPERATURE ABOVE 175°F

TABLE 4.4-5

REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE

SPECIMEN	REMOVAL INTERVAL
1.	8 years
2.	16 years
3.	23 years
4.	30 years
5.	35 years
6.	40 years

PORV BLOCK VALVES

LIMITING CONDITION FOR OPERATION

3.4.12 Each Power Operated Relief Valve (PORV) Block Valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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.

SURVEILLANCE REQUIREMENTS

4.4.12 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel.

---

* Until October 1, 1981, in lieu of closing and removing power to the block valve V-1403, the PORV, V-1402, may be deenergized in the closed position such that it is incapable of being opened.

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 200°F and

2. During heatup and isothermal conditions when the temperature of any RCS cold leg is less than or equal to 180°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 200°F and less than or equal to 334°F.

2. During heatup and isothermal conditions when the temperature of any RCS cold leg is greater than or equal to 180°F and less than or equal to 334°F.

APPLICABILITY: MODES 3#, 4 and 5*.

ACTION:

a. With less than two PORVs OPERABLE and while at Hot Standby 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 334°F.

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

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 3#, 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 Specificsation 3.4.9.1.

SURVEILLANCE REQUIREMENTS

4.4.14 Prior to starting a reactor cooalnt 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 334°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 253°F a maximum of only one high pressure safety injection pump is to be OPERABLE with its associated header stop valves open.

c. Prior to decreasing the reactor coolant system temperature below 220°F, all high pressure safety injection pumps will 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 253°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 235°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.

3/4.1.2 BORATION SYSTEMS (Continued)

The boron addition capability after the plant has been placed in MODES 5 and 6 requires either 1660 gallons of 8% boric acid solution from the boric acid tanks or 1630 gallons of 1720 ppm borated water from the refueling water tank to makeup for contraction of the primary coolant that could occur if the temperature is lowered from 200°F to 140°F.

The restrictions associated with the establishing of the flow path from the RWT to the RCS via a single HPSI pump provide assurance that Appendix G pressure/temperature limits will not be exceeded in the case of any inadvertent pressure transient due to a mass addition to the RCS.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of a CEA ejection accident are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met.

The ACTION statements applicable to an immovable or untrippable CEA and to a large misalignment ($\geq$ 15 inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (< 15 inches) of the CEAs, there is 1) a small degradation in the peaking factors relative to those assumed in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 2) a small effect on the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 3) a small effect on the available SHUTDOWN MARGIN, and 4) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the small misalignment of a CEA permits a one hour time interval during which attempts may be made to restore the CEA to within its alignment requirements prior to initiating a reduction in THERMAL POWER. The one hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs, and (3) minimize the effects of xenon redistribution.

Overpower margin is provided to protect the core in the event of a large misalignment ($\geq$ 15 inches) of a CEA. However, this misalignment would cause distortion of the core power distribution. This distribution may, in turn, have a significant effect on (1) the available SHUTDOWN MARGIN, (2) the time-dependent long-term power distributions relative to those used in generating LCOs and LSSS setpoints, and (3) the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the large misalignment of the CEA requires a prompt realignment of the misaligned CEA.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES (Continued)

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements brings the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in 1) local burnup, 2) peaking factors, and 3) available shutdown margin which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

The requirement to reduce power in certain time limits, depending upon the previous F_r^t, is to eliminate a potential nonconservatism for situations when a CEA has been declared inoperable. A worst case analysis has shown that a DNBR SAFDL violation may occur during the second hour after the CEA misalignment if this requirement is not met. This potential DNBR SAFDL violation is eliminated by limiting the time operation is permitted at FULL POWER before power reductions are required. These reductions will be necessary once the deviated CEA has been declared inoperable. The time allowed to continue operation at a reduced power level can be permitted for the following reasons:

1. The margin calculations that support the Technical Specifications are based on a steady-state radial peak of $F^t > 1.70$.

2. When the actual $F_r^t \leq 1.70$, significant additional margin exists.

3. This additional margin can be credited to offset the increase in F_r^t with time that can occur following a CEA misalignment.

4. This increase in F_r^t is caused by xenon redistribution.

5. The present analysis can support allowing a misalignment to exist for up to 60 minutes without correction, if the initial $F_r^t \leq 1.67$.

Operability of the CEA position indicators (Specification 3.1.3.3) is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits and ensures proper operation of the rod block circuit. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

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 the DNBR limit 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 restrictions on starting a Reactor Coolant Pump are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the Reactor Coolant Pumps to when the secondary water temperature of each steam generator is less than 30°F above each of the Reactor Coolant System cold leg temperatures.

3/4.4.2 and 3/4.4.3 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 2×10^5 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.

3/4.4.2 and 3/4.4.3 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 Reactor Coolant 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 Code, 1974 Edition.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valve against water relief. The power operated relief valve and steam bubble function to relieve RCS pressure during all design transients. Operation of the power operated relief valve 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 required pressurizer heater capacity is capable of maintaining natural circulation subcooling. Operability of the heaters, which are powered by a diesel generator bus, ensures ability to maintain pressure control even with loss of offsite power.

3/4.4.5 STEAM GENERATORS

One OPERABLE steam generator provides sufficient heat removal capability to remove decay heat after a reactor shutdown. The requirement for two steam generators capable of removing decay heat, combined with the requirements of Specifications 3.7.1.1, 3.7.1.2 and 3.7.1.3 ensures adequate decay heat removal capabilities for RCS temperatures greater than 325°F if one steam generator becomes inoperable due to single failure considerations. Below 325°F, decay heat is removed by the shutdown cooling system.

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 demage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. 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.

time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM and a concurrent loss of offsite electrical power. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the St. Lucie site, such as site boundary location and meteorological conditions, were not considered in this evaluation. The NRC is finalizing site specific criteria which will be used as the basis for the reevaluation of the specific activity limits of this site. This reevaluation may result in higher limits.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity > 1.0 μCi/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 μCi/gram DOSE EQUIVALENT I131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 10 percent of the unit's yearly operating time since the activity levels allowed by Figure 3.4-1 increase the 2 hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture.

Reducing T_{avg} to < 500°F prevents the release of activity should a steam generator tube rupture since the saturaction pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take correction action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

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 5.2.1 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 are tensile at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside surface 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 than at the outside surface location, the inside surface flaw may be more limiting. Consequently, for the heatup analysis, both the inside surface 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 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.

Since neutron irradiation damage is also greater at the inside surface, the inside surface flaw location is the limiting location during cooldown. Consequently, only the inside surface flaw must be evaluated for the cooldown analysis.

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 possible errors in the pressure and temperature sensing instruments.

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, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irratiation 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 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 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 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	A508C1.2	-	-	.008	+20	+70	+90	+30	133	86
Bottom Head Plate	C-10-1	A533BC1.1	-	-	.010	-40	+42	+62	+2	120	78
Bottom Head Plate	C-9-2	A533BC1.1	-	-	.011	-40	-18	+2	-40	146	95
Bottom Head Plate	C-9-3	A533BC1.1	-	-	.013	-70	-20	0	-60	148	96
Bottom Head Plate	C-9-1	A533BC1.1	-	-	.011	-30	+10	+30	-30	138	90
Inlet Nozzle	C-4-3	A508C1.2	-	-	.005	0	0	+20	0	111	72
Inlet Nozzle	C-4-2	A508C1.2	-	-	.004	0	+20	+40	0	146	95
Inlet Nozzle	C-4-1	A508C1.2	-	-	.005	+10	-25	-5	10	144	94
Inlet Nozzle	C-4-4	A508C1.2	-	-	.004	0	+10	+30	0	139	90
Inlet Nozzle Ext.	C-16-3	A508C1.2	-	-	.001	+10	+52	+72	+12	139	90
Inlet Nozzle Ext.	C-16-2	A508C1.2	-	-	.011	+10	+52	+72	+12	139	90
Inlet Nozzle Ext.	C-16-1	A508C1.2	-	-	.011	+10	+52	+72	+12	139	90
Inlet Nozzle Ext.	C-16-4	A508C1.2	-	-	.011	+10	+52	+72	+12	139	90

TABLE B 3/4.4-1 (Cont'd)

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)
Outlet Nozzle	C-3-1	A508C1.2	-	-	.009	+10	+88	+108	+48	119	77
Outlet Nozzle	C-3-2	A508C1.2	-	-	.010	-20	+92	+112	+52	111	72
Outlet Nozzle Ext.	C-17-1	A508C1.2	-	-	.013	+20	-	-	+28(5)	126	82
Outlet Nozzle Ext.	C-17-2	A508C1.2	-	-	.013	+20	-	-	+28(5)	126	82
Upper Shell Plate	C-6-3	A533BC1.1	-	-	.011	-10	+30	+50	-10	129	84
Upper Shell Plate	C-6-2	A533BC1.1	-	-	.010	-30	+45	+65	+5	123	80
Upper Shell Plate	C-6-1	A533BC1.1	-	-	.012	+10	+42	+62	+10	105	68
Inter. Shell Plate	C-7-1	A533BC1.1	0.11	0.64	0.004	0	+26	+46	0	126	82
Inter. Shell Plate	C-7-2	A533BC1.1	0.11	0.64	0.004	-30	+30	+50	-10	131	85
Inter. Shell Plate	C-7-3	A533BC1.1	0.11	0.58	0.004	-30	+50	+70	+10	117	76
Lower Shell Plate	C-8-3	A533BC1.1	0.12	0.58	0.004	0	+26	+46	0	136	88
Lower Shell Plate	C-8-1	A533BC1.1	0.15	0.56	0.006	-10	+60	+80	+20	126	82
Lower Shell Plate	C-8-2	A533BC1.1	0.15	0.57	0.006	0	+32	+52	20(7)	120	78
Closure Head Flange	C-2	A508C1.2	-	-	.008	+20	-	-	+20(5)	143	93
Closure Head Peels	C-21-2	A533BC1.1	-	-	.012	-30	+40	+60	0	133	86
Closure Head Peels	C-21-2	A533BC1.1	-	-	.012	-30	+40	+60	0	133	86

TABLE B 3/4.4-1 (Cont'd)

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)
Closure Head Peels	C-21-1	A533BC1.1	-	-	.013	-10	0	+20	-10	138	90
Closure Head Peels	C-21-1	A533BC1.1	-	-	.013	-10	0	+20	-10	138	90
Closure Head Peels	C-21-2	A533BC1.1	-	-	.012	-30	+40	+60	0	133	86
Closure Head Peels	C-21-3	A533BC1.1	-	-	.013	-40	+36	+56	-4	129	84
Closure Head Dome	C-20-1	A533BC1.1	-	-	.014	-10	+44	+64	+4	105	68
Inter. Shell Long. Welds	2-203 A,B,C	A533B	0.12	0.20	.018	-	-	-	-56(6)	-	-
Lower Shell Long. Welds	3-203 A,B,C	A533B	0.30	0.64	.013	-	-	-	-56(6)	-	-
Lower-to-Inter. Shell Seam Weld	9-203	A533B	0.23	0.11	.013	-60(7)	-	-36(7)	-60(7)	-	144(7)

NOTES:
(1) Charpy 50 ft-lb and 35 mils lateral expansion index temperature (lower bound)
(2) Determined using Branch Technical Position MTEB 5-2, Section 1.1(3)(b)
(3) Determined by using Branch Technical Position MTEB 5-2 Section 1.2
(4) As per ASME B&PV Code, Section III, NB-2331
(5) Charpy test data either do not have lateral expansion value or the data are not legible. The reference temperature from Charpy test data was obtained by following MTEB Position 5.2, Section 1.1(4)
(6) Estimated based on generic data for C-E submerged arc welds ("Evaluation of Pressurized Thermal Shock Effects due to Small Break LOCA's with Loss of Feedwater for the Combustion Engineering NSSS," CEN-189, December 1981).
(7) Surveillance Program Data

DELETED

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 number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

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.

3/4.4.10 STRUCTURAL INTEGRITY

The required inspection programs for the Reactor Coolant System components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for the Reactor Coolant System components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code "Inservice Inspection of Nuclear Reactor Coolant Systems", 1971 Edition, and Addenda through Winter 1972.

All areas scheduled for volumetric examination have been pre-service mapped using equipment, techniques and procedures anticipated for use during post-operation examinations. To assure that consideration is given to the use of new or improved inspection equipment, techniques and procedures, the Inservice Inspection Program will be periodically reviewed on a 5 year basis.

The use of conventional nondestructive, direct visual and remote visual test techniques can be applied to the inspection of most reactor coolant loop components except the reactor vessel. The reactor vessel requires special consideration because of the radiation levels and the requirement for remote underwater accessibility.

The techniques anticipated for inservice inspection include visual inspections, ultrasonic, radiographic, magnetic particle and dye penetrant testing of selected parts.

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 below 334°F. 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.

3/4.5.1 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 $\leq$ 253°F and $\leq$ 220°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. 81

TO FACILITY OPERATING LICENSE NO. DPR-67

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT, UNIT NO. 1

DOCKET NO. 50-335

INTRODUCTION

The St. Lucie, Unit 1 design uses power operated relief valves (PORVs) for overpressure protection during low temperature operating conditions. Pressure/temperature (P/T) limits for plant cooldown and heatup are specified in Figures 3.4-2a, 3.4-2b and 3.4-2c of the Technical Specifications (TS). The licensee's analyses documented in Appendix 5B of FSAR include (1) a postulated inadvertent reactor coolant pump (RCP) start event, assuming a maximum secondary to primary temperature differential of 100°F, and (2) a postulated inadvertent safety injection actuation event, assuming two high pressure safety injection (HPSI) pumps and three charging pumps injecting water into the reactor coolant system (RCS) for temperatures above 195°F, one HPSI pump and three charging pumps injecting water into the RCS for temperatures below 195°F with one HPSI pump disabled and three charging pumps injecting water into the RCS for temperatures below 155°F with both HPSI pumps disabled. All the above analyses performed to support LTOP system assume only one PORV operating during the overpressure transients.

The licensee indicated that the removal of the St. Lucie Unit 1 thermal shield resulted in a reduction in the period of applicability of the current TS for the RCS P/T limitations from 10 years of full power operation to 7.4 Effective Full Power Years (EFPY). As a result, the existing St. Lucie Unit 1 LTOP system will be inadequate for RCS pressure boundary protection at 7.4 EFPY.

By letter dated March 17, 1987, the licensee submitted proposed TS changes and the supporting analyses to ensure that the RCS pressure boundary integrity will be maintained in the low temperature modes of operation during the period from 7.4 EFPY to 10 EFPY. In its submittal, the licensee provided new P/T limits and a number of administrative and system modifications to the existing LTOP system have been identified. The licensee proposes to change TS Sections 1.16, 4.1.2.1, 4.1.2.3, 3.4.1.4.1, 3.4.9.1, 4.4.9.1, 3.4.13, 4.4.13, 3.4.14, 4.4.14, 3.5.3, 3/4.1.2, 3/4.4.1, 3/4.4.9, 3/4.4.13, 3/4.5.2, and 3/4.5.3 to support the proposed changes to the P/T limits and the LTOP system.

EVALUATION - LOW TEMPERATURE OVERPRESSURE PROTECTION

In the design of the proposed LTOP system, the licensee considered the following overpressurization events during plant heatup and cooldown conditions:

a) Mass Addition Events

1) Actuation of two HPSI pumps and all three charging pumps,

2) Actuation of one HPSI pump and all three charging pumps when one HPSI pump is disabled,

3) Actuation of all three charging pumps when all HPSI pumps are disabled, and

4) Actuation of one HPSI pump when all three charging pumps and the other HPSI pumps are disabled.

b) Energy Addition Events

1) Reactor Coolant Pump (RCP) start, with a positive secondary to primary temperature differential,

2) Decay heat addition due to shutdown cooling system (SDCS) isolation, and

3) Inadvertent pressurizer heater input.

Based on the results of a pressure transient analysis for the above events and the P/T limits presented in the revised Figures 3.4-2a, 3.4-2b and 3.4-3, the licensee, in its letter dated March 17, 1987, proposed system modifications and administrative controls as follows:

a) System Modifications

1) The existing setpoint of 465 psia is increased to 530 psia for both PORVs.

2) A second low-pressure setpoint of 350 psia is added to both PORVs.

As a result of these modifications, both PORVs will have two setpoints for LTOP. Pressure transient analysis only takes credit for one PORV for event mitigation.

b) Administrative Controls

1) The upper temperature limit for the LTOP mode of operation is increased from the current value of 275°F to a new value of 334°F.

2) The PORV setpoint will be at 350 psia during heatup when RCS cold leg temperature (Tc) is less than 180°F and during cooldown when Tc is less than 200°F. The PORV setpoint will be at 530 psia during heatup when Tc is between 180°F and 334°F and during cooldown when Tc is between 200°F and 334°F.

3) The maximum secondary to primary temperature differential for RCS pump starts is reduced from the current 45°F to a new value of 30°F.

4) The minimum RCS temperature above which the two HPSI pumps shall be operable is increased from the current value of 215°F to a new value of 253°F. Below 253°F, one HPSI pump shall be disabled.

5) The minimum RCS temperature below which all HPSI pumps must be disabled is increased from the current value of 165°F to a new value of 220°F.

6) The flow path from the Refueling Water Storage Tank (RWST) to the RCS via a HPSI pump shall only be established if a concern of the RCS pressure boundary integrity does not exist or if no charging pump is operable in which case all three charging pumps shall be disabled and heatup and cooldown rates shall be further restricted.

The licensee's above listed proposed modifications have been reflected in its proposed TS changes given in the introduction section of this report. The staff has reviewed the results of the licensee's reanalysis of pressure transients and its proposed TS changes and concludes that the proposed changes are adequate to prevent violation of the Appendix G P/T limits during an operating period ending at 10 EFPY.

The staff accepts the design bases for the LTOP system with regard to its precluding some overpressure events by the use of administrative controls. Since this approach has been used in its previous analyses documented in Appendix 5B of St. Lucie Unit 1 FSAR, the design also meets the intent of Section B.2 of the Branch Technical Position RSB 5-2.

EVALUATION - RCS PRESSURE-TEMPERATURE LIMIT FIGURES

Pressure-temperature limits must be calculated in accordance with the requirements of Appendix G, 10 CFR 50, which became effective on July 26, 1983. Pressure-temperature limits that are calculated in accordance with the requirements of Appendix G, 10 CFR 50, are dependent upon the initial reference temperature (RT_{NDT}) for the limiting materials in the beltline and closure flange regions of the reactor vessel and the increase in RT_{NDT} resulting from neutron irradiation damage to the limiting beltline material. The St. Lucie Unit 1 reactor vessel was procured to the ASME Code requirements which did specify fracture toughness testing to determine the initial RT_{NDT} for each vessel material. In the Bases Section of the TS, the licensee indicated that the initial RT_{NDT} for the limiting material in the closure flange region of the vessel was estimated using the method recommended by the staff in NRC Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements," which is documented in the Standard Review Plan, Section 5.3.2, "Pressure-Temperature Limits." The initial RT_{NDT} for the beltline weld metal was estimated based on generic weld metal data for Combustion Engineering (CE) submerged arc welds, which is reported in CE Report CEN-189, "Evaluation of Pressurized Thermal Shock Effects Due to Small Break LOCA's with Loss of Feedwater for the Combustion Engineering NSSS," December 1981. The St. Lucie Unit 1 reactor vessel was fabricated by Combustion Engineering. Hence, the generic values of initial RT_{NDT} would be applicable for the St. Lucie Unit 1 weld metal.

The limiting beltline material is the weld metal which was fabricated using weld wire (heat no. 34B009) and Linde 0091 flux (lot no. 3889). The licensee indicates that the generic data results in an initial RT_{NDT} of -56°F for this material. The chemical composition of this material is given in Table B3/4.4-1 of the TS. The licensee indicates that the limiting closure flange region material is the vessel flange, in which the initial RT_{NDT} is estimated as 30°F.

The increase in RT_{NDT} resulting from neutron irradiation damage depends upon the predicted amount of neutron fluence and the rate of embrittlement of the limiting reactor vessel beltline material. The licensee estimated that the neutron fluence at the inside surface of the limiting weld will be 0.78×10^{19} n/cm^2 at 10 EFPY.

The increase in RT_{NDT} resulting from neutron irradiation damage was estimated according to Draft Regulatory Guide 1.99 Rev. 2 "Radiation Embrittlement of Reactor Vessel Materials." Table 1 compares the observed increase in RT_{NDT} of the surveillance material test data from St. Lucie Capsule W-97. The surveillance data was submitted for staff review in a letter from J.W. Williams, Jr. to D.G. Eisenhut dated December 14, 1983. The surveillance material test results indicate that the increase RT_{NDT} of surveillance material is significantly less than that predicted by Regulatory Guide 1.99 Rev. 2. Hence, for the St. Lucie Unit 1 limiting reactor vessel beltline materials, the Regulatory Guide should provide a conservative estimate as to the amount of increase in RT_{NDT} resulting from neutron irradiation.

We have used the unirradiated RT_{NDT} for beltline and closure flange materials, which were previously discussed, the neutron fluence estimates of the licensee, the Regulatory Guide 1.99 Rev. 2 method of estimating neutron irradiation damage, and Standard Review Plan 5.3.2 method of calculating pressure-temperature limits to evaluate the licensee's proposed pressure-temperature limits. Our evaluation indicates that the proposed pressure-temperature limits meet the safety margins of Appendix G, 10 CFR 50, for a period of time corresponding to 10 EFPY. Hence, the proposed pressure-temperature limit may be incorporated into the Technical Specifications for St. Lucie Unit 1.

SUMMARY

Based on the above evaluation, the staff concludes that the licensee's proposed changes to TS Sections 1.16, 4.1.2.1, 4.1.2.3, 3.4.1.4.1, 3.4.9.1, 4.4.9.1, 3.4.13, 4.4.13, 3.4.14, 4.4.14, 3.5.3, 3/4.1.2, 3/4.4.1, 3/4.4.9, 3/4.4.13, 3/4.5.2, and 3/4.5.3 with its proposed modifications as submitted in its letter dated March 17, 1987 are acceptable. The proposed LTOP system is designed in accordance with the requirements set forth in the Branch Technical Position RSB 5-2 and it is adequate to prevent violation of the Appendix G P/T limits during an operating period ending at 10 EFPY. Implementation of the proposed LTOP system will not result in a reduction in the margin of safety presently afforded by TS. Therefore, the staff concludes that the licensee's proposed modifications of system design, administrative controls, and their associated TS changes are acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or a change in a surveillance requirement. The staff has determined that the 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 published a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the 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 the 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.

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Principal Contributors:

C. Liang
B. Elliot
E. Tourigny

Table 1

Comparison of Measured and Calculated Increase in Reference Temperature, RT_{NDT} for Material from Surveillance Capsule W-97

Material	Neutron Fluence (E>1MeV)	Measured Increase in RT_{NDT}	*Calculated Increase in RT_{NDT} Using R.G.1.99 Rev. 2
	(n/cm^2)	(°F)	(°F)
Base Metal (Long. Specimens)	5.5×10^{18}	68	154
Base Metal (Trans. Specimens)	5.5×10^{18}	70	154
Weld Metal	5.5×10^{18}	74	156

*Mean Value