December 14, 1978

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Docket No. 50-335

Dr. Robert E. Uhrig
Vice President, Advanced
Systems & Technology
Florida Power & Light Company
P. O. Box 529100
Miami, Florida 33152

Dear Dr. Uhrig:

The Commission has issued the enclosed Amendment No. 28 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 requests dated February 24 and 27, 1978.

This amendment revises the Technical Specifications to increase the minimum required volume of water in the Refueling Water Tank (RWT) and to allow an increase in the coolant temperature at which shutdown cooling is initiated.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Amendment No. 28 to DPR-67
2. Safety Evaluation
3. Notice

cc w/enclosures: see next page

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NRC FORM 318 (9-76) NRCM 0240 ☆ U.S. GOVERNMENT PRINTING OFFICE: 1978 - 265 - 769

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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. 28
License No. DPR-67

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

   A. The applications for amendment by Florida Power & Light Company (the licensee) dated February 24 and 27, 1978, comply 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 applications, 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, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-67 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 28, 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

Gerald B. Zwetzig
for Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 14, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 28

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.

Pages

V
1-7
3/4 1-18
3/4 4-23a
3/4 4-23b
3/4 4-23c
3/4 5-3
3/4 5-7
3/4 5-8
B 3/4 1-2
B 3/4 4-2
B 3/4 7-2

INDEX

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

TABLE 1.1

OPERATIONAL MODES

MODE		REACTIVITY CONDITION, K_{eff}	%RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1.	POWER OPERATION	≥ 0.99	$> 5\%$	$\geq 325°F$
2.	STARTUP	≥ 0.99	$\leq 5\%$	$\geq 325°F$
3.	HOT STANDBY	< 0.99	0	$\geq 325°F$
4.	HOT SHUTDOWN	< 0.99	0	$325°F > T_{avg} > 200°F$
5.	COLD SHUTDOWN	< 0.99	0	$\leq 200°F$
6.	REFUELING**	≤ 0.95	0	$\leq 140°F$

* Excluding decay heat.
**Reactor vessel head unbolted or removed and fuel in the vessel.

TABLE 1.2

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 6 months.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
N.A.	Not applicable.

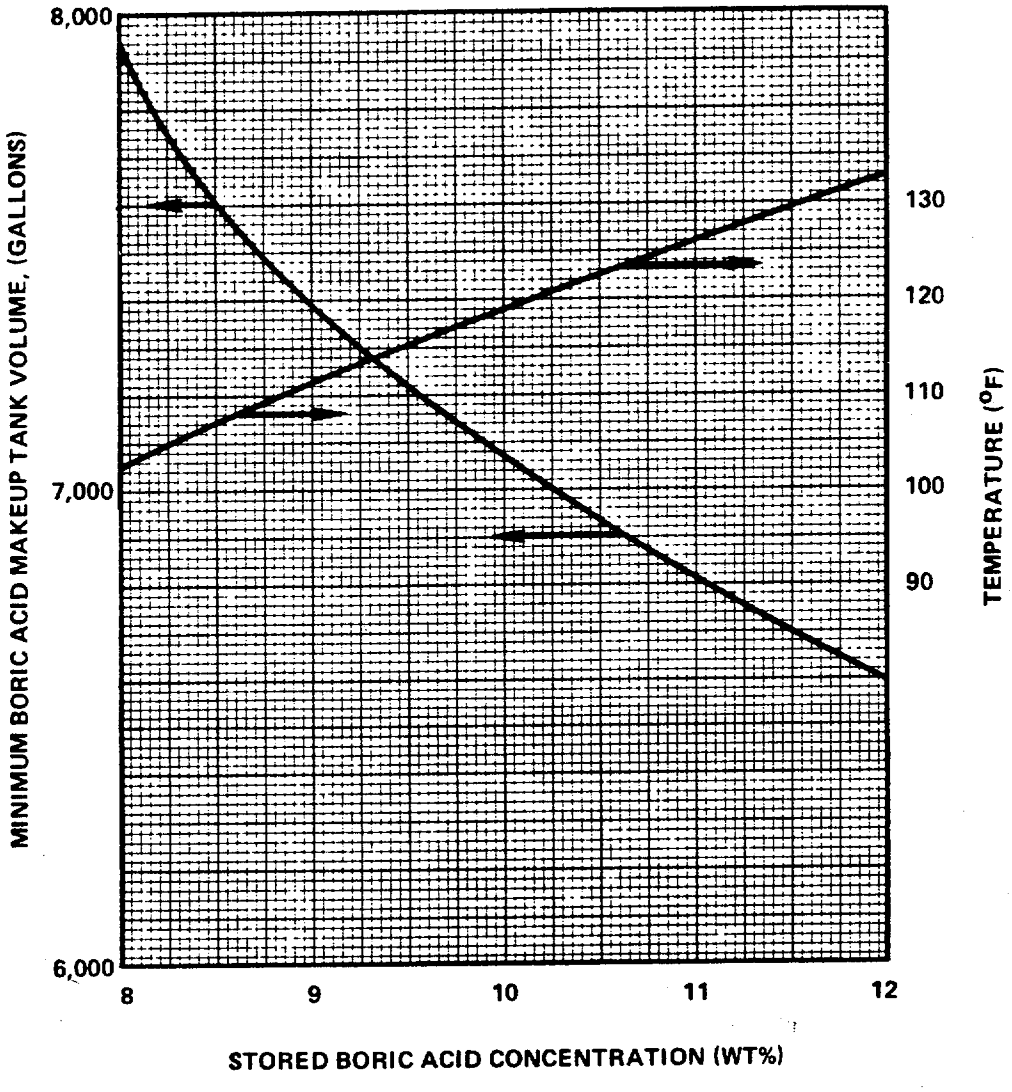

Figure 3.1-1 Minimum Boric Acid Makeup Tank Volume and Temperature as a Function of Stored Boric Acid Concentration

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 At least two of the following three borated water sources shall be OPERABLE:

a. Two boric acid makeup tanks and one associated heat tracing circuit with the contents of the tanks in accordance with Figure 3.1-1, and

b. The refueling water tank with:

1. A minimum contained volume of 401,800 gallons of water,

2. A minimum boron concentration of 1720 ppm,

3. A maximum solution temperature of 100°F,

4. A minimum solution temperature of 55°F when in MODES 1 and 2, and

5. A minimum solution temperature of 40°F when in MODES 3 and 4.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one borated water source OPERABLE, restore at least two borated water sources 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 1% $\Delta k/k$ at 200°F; restore at least two borated water sources to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 At least two borated water sources shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in each water source,

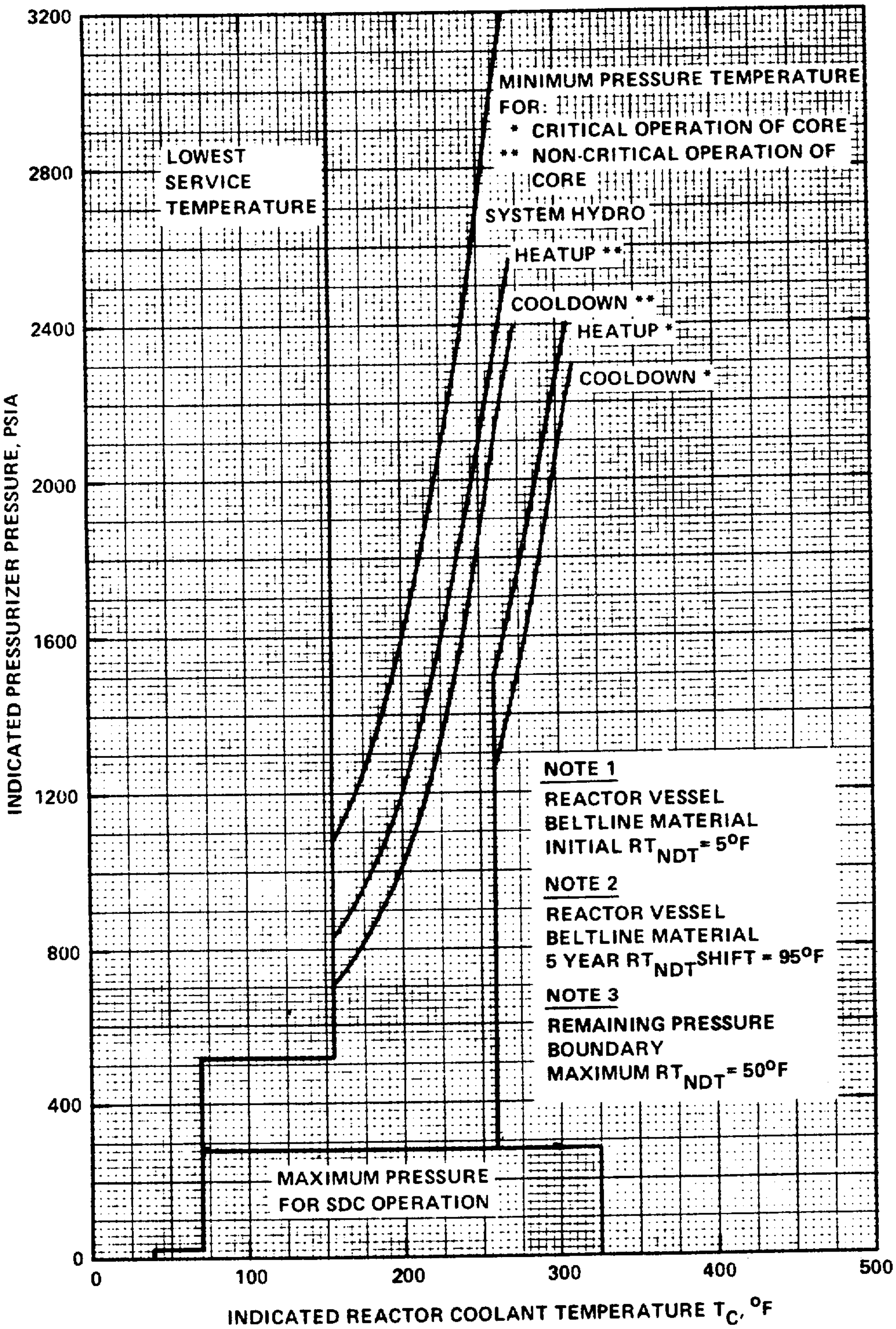

FIGURE 3.4-2a

Reactor Coolant System Pressure Temperature Limitations
for up to 5 Years of Full Power Operation

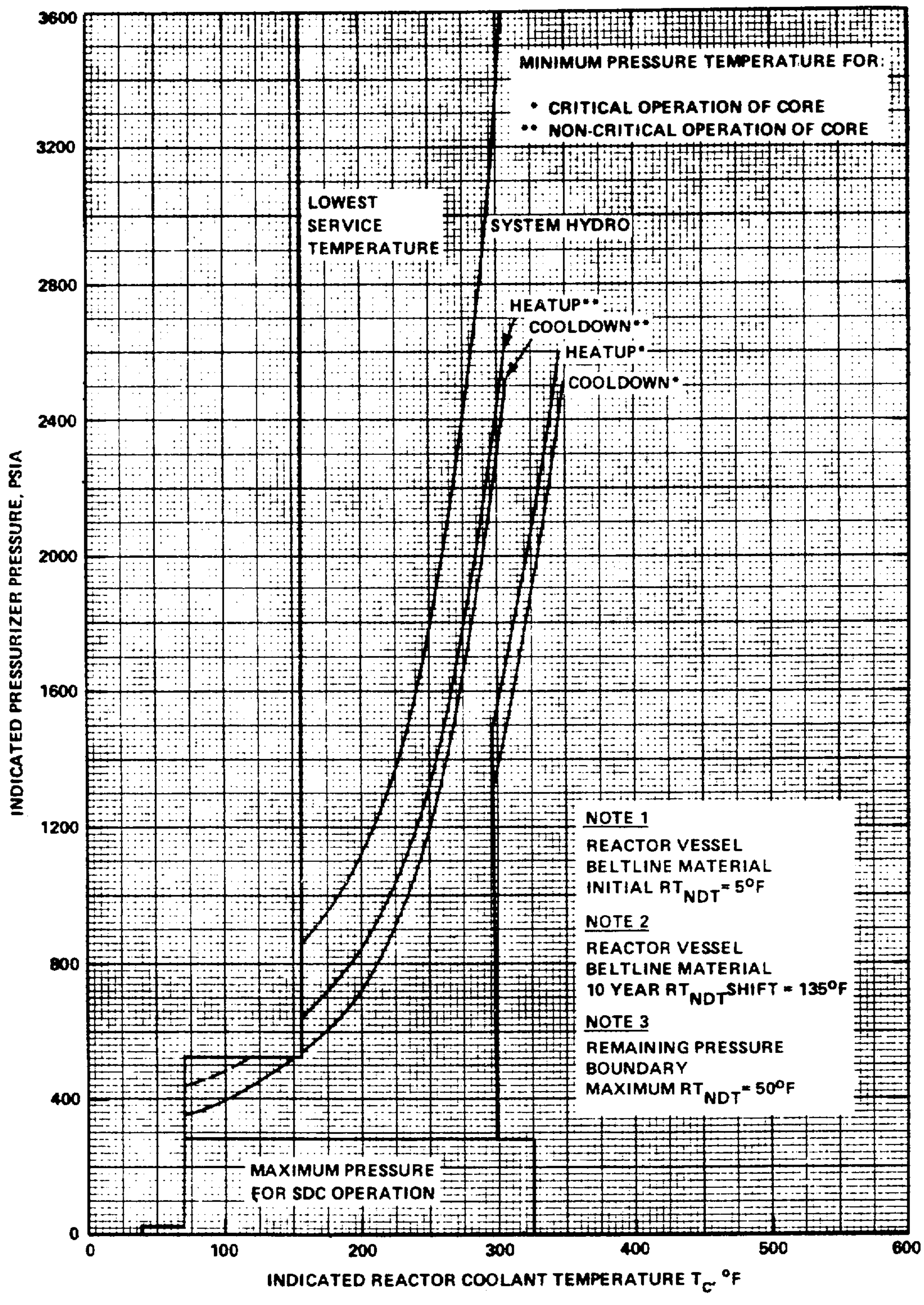

FIGURE 3.4-2b

Reactor Coolant System Pressure Temperature Limitations for up to 10 Years of Full Power Operation

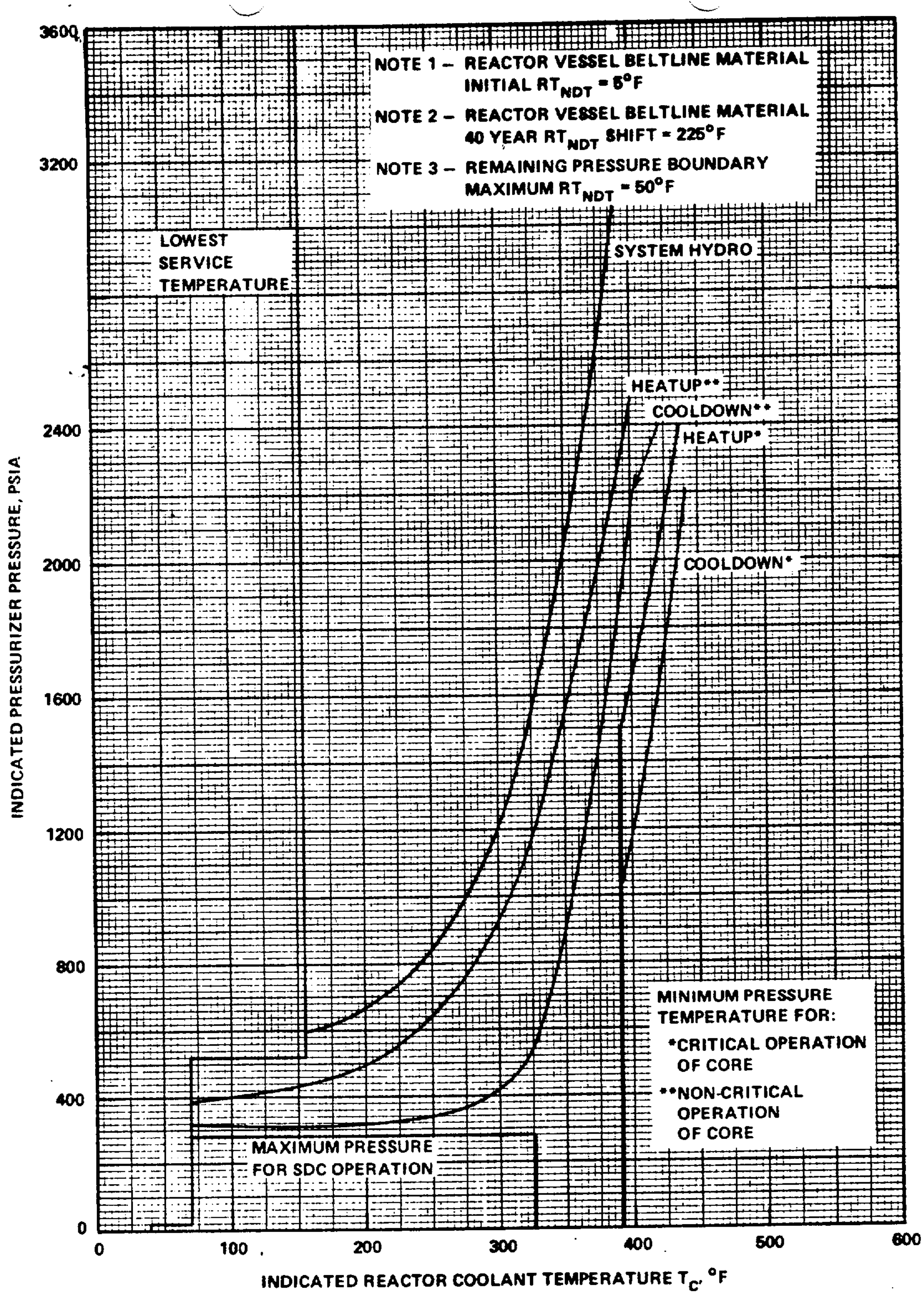

FIGURE 3.4-2c

Reactor Coolant System Pressure Temperature Limitations for up to 40 Years of Full Power Operation

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

ECCS SUBSYSTEMS - $T_{avg} \geq 325°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high-pressure safety injection (HPSI) pump (one ECCS subsystem shall include HPSI pump A and the second ECCS subsystem shall include either HPSI pump B or C),

b. One OPERABLE low-pressure safety injection pump, and

c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. 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.

*With pressurizer pressure $\geq$ 1750 psia.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
1. V-3659	1. Mini-flow isolation	1. Open
2. V-3660	2. Mini-flow isolation	2. Open

b. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying that each high-pressure safety injection pump:

   a) Starts (unless already operating) from the control room.

   b) Develops a discharge pressure of $\geq$ 1138 psig on recirculation flow.

   c) Operates for at least 15 minutes.

2. Verifying that each low-pressure safety injection pump:

   a) Starts (unless already operating) from the control room.

   b) Develops a discharge pressure of $\geq$ 175 psig on recirculation flow.

   c) Operates for at least 15 minutes.

3. Verifying that upon a recirculation actuation signal, the containment sump isolation valves open.

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

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. One OPERABLE high-pressure safety injection pump, and

b. An OPERABLE flow path capable of taking suction from the refueling water tank on a safety injection actuation signal and automatically transferring suction to the containment sump on a recirculation actuation signal.

APPLICABILITY: MODES 3* and 4.

ACTION:

a. With no ECCS subsystem OPERABLE, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

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

*With pressurizer pressure < 1750 psia.

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.

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 3.3% $\Delta k/k$ is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN required by Specification 3.1.1.1 is based upon this limiting condition and is consistent with FSAR accident analysis assumptions. For earlier periods during the fuel cycle, this value is conservative. With $T_{avg} \leq 200°F$, the reactivity transients resulting from any postulated accident are minimal and a 1% $\Delta k/k$ shutdown margin provides adequate protection.

3/4.1.1.3 BORON DILUTION AND ADDITION

A minimum flow rate of at least 3000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration changes in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 11,400 cubic feet in approximately 26 minutes. The reactivity change rate associated with boron concentration changes will be within the capability for operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limiting values assumed for the MTC used in the accident and transient analyses were $+ 0.5 \times 10^{-4}$ $\Delta k/k/°F$ for THERMAL POWER levels $<$ 70% of RATED THERMAL POWER, $+ 0.2 \times 10^{-4}$ $\Delta k/k/°F$ for THERMAL POWER levels $>$ 70% of RATED THERMAL and $- 2.2 \times 10^{-4}$ $\Delta k/k/°F$ at RATED THERMAL POWER. Therefore, these limiting values are included in this specification. Determination of MTC at the specified conditions ensures that the maximum positive and/or negative values of the MTC will not exceed the limiting values.

BASES

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

The MTC is expected to be slightly negative at operating conditions. However, at the beginning of the fuel cycle, the MTC may be slightly positive at operating conditions and since it will become more positive at lower temperatures, this specification is provided to restrict reactor operation when T_{avg} is significantly below the normal operating temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from all operating conditions of 1.0% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 7,925 gallons of 8.0% boric acid solution from the boric acid tanks or 13,700 gallons of 1720 ppm borated water from the refueling water tank.

The requirements for a minimum contained volume of 401,800 gallons of borated water in the refueling water tank ensures the capability for borating the RCS to the desired level. The specified quantity of borated water is consistent with the ECCS requirements of Specification 3.5.4. Therefore, the larger volume of borated water is specified here too.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients. STARTUP and POWER OPERATION may be initiated and may proceed with one or two reactor coolant pumps not in operation after the setpoints for the Power Level-High, Reactor Coolant Flow-Low, and Thermal Margin/Low Pressure trips have been reduced to their specified values. Reducing these trip setpoints ensures that the DNBR will be maintained above 1.30 during three pump operation and that during two pump operation the core void fraction will be limited to ensure parallel channel flow stability within the core and thereby prevent premature DNB.

A single reactor coolant loop with its steam generator filled above the low level trip setpoint provides sufficient heat removal capability for core cooling while in MODES 2 and 3; however, single failure considerations require plant cooldown if component repairs and/or corrective actions cannot be made within the allowable out-of-service time.

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.

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.

SAFETY VALVES (Continued)

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

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 damage 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.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within its design pressure of 1025 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition and ASME Code for Pumps and Valves, Class II. The total relieving capacity for all valves on all of the steam lines is 11.91×10^6 lbs/hr which is 106.7 percent the total secondary steam flow of 11.17×10^6 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for removing decay heat.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Level-High channels. The reactor trip setpoint reductions are derived on the following bases:

For two loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (106.5)$$

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line

106.5 = Power Level-High Trip Setpoint for two loop operation

X = Total relieving capacity of all safety valves per steam line in lbs/hour (5.95×10^6 lbs/hr.)

Y = Maximum relieving capacity of any one safety valve in lbs/hour (7.44×10^5 lbs/hr.)

3/4.7.1.2 AUXILIARY FEEDWATER PUMPS

The OPERABILITY of the auxiliary feedwater pumps ensures that the Reactor Coolant System can be cooled down to less than 325°F from normal operating conditions in the event of a total loss of off-site power.

Any two of the three auxiliary feedwater pumps have the required capacity to provide sufficient feedwater flow to remove reactor decay heat and reduce the RCS temperature to 325°F where the shutdown cooling system may be placed into operation for continued cooldown.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available for cooldown of the Reactor Coolant System to less than 325°F in the event of a total loss of off-site power. The minimum water volume is sufficient to maintain the RCS at HOT STANDBY conditions for 8 hours with steam discharge to atmosphere.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. The dose calculations for an assumed steam line rupture include the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line and a concurrent loss of offsite electrical power. These values are consistent with the assumptions used in the accident analyses.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 28 TO FACILITY OPERATING LICENSE NO. DPR-67

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT, UNIT NO. 1

DOCKET NO. 50-335

Introduction

By applications dated February 24 and 27, 1978, Florida Power and Light Company (FPL) requested amendments to the Technical Specifications for the St. Lucie Plant, Unit No. 1. The amendments would change the Technical Specifications to increase the minimum required volume of water in the Refueling Water Tank (RWT) and allow an increase in the coolant temperature at which shutdown cooling is initiated. We have combined these two proposed amendments in this action.

Discussion and Evaluation

Refueling Water Tank Level

This proposed change to the Technical Specifications (Reference 1) consists of increasing the minimum required volume of water in the Refueling Water Tank (RWT) from 371,800 gallons to 401,800 gallons. (The RWT is described in the St. Lucie Unit No. 1 Final Safety Analysis Report (FSAR) (Reference 2)). FPL requests this change in order to assure that, in the event of a Loss-of-Coolant-Accident (LOCA), 305,600 gallons of borated water would be available in the RWT to be delivered to the containment by the emergency core cooling pumps and 66,200 gallons would remain in the tank and would provide adequate net positive suction head (NPSH) for the pumps.

Recently, FPL identified (Reference 3) a portion of the piping between the RWT and charging pumps, downstream from a manually operated valve, as non-seismic Class I piping. Should the line rupture, assuming the valve was aligned in an open position, it was postulated that a portion of the RWT inventory could be lost. FPL has determined that in order to account for the loss of water through the break during a conservatively estimated injection phase (~57 minutes), the minimum volume of the water in the RWT should be increased by approximately 10 percent up to a volume of 401,800 gallons.

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We have verified FPL's assertions and conclude that the increased water inventory in the RWT would compensate for the water lost through the postulated break and that the safety injection pumps would be able to deliver the required water volume to the containment. We have also considered the effect of increased borated water inventory on boric acid precipitation during post-LOCA long term cooling and conclude that the required switchover time from cold to hot leg injection would not change significantly.

After reviewing the information provided by FPL and making independent calculations, we have determined that the proposed increase in the minimum RWT water inventory will provide the required improvement in the performance of the emergency core cooling system.

Temperature for Shutdown Cooling

The proposed change in the Technical Specifications (Reference 4) consists of changing the upper temperature limit for the initiation of shutdown cooling from 300°F to 325°F. FPL requests this change in order to be able to satisfy the criteria of FSAR Section 10.3.2 for cooldown to the shutdown cooling window in the event that the offsite power is unavailable for an extended period of time. On April 8, 1977, FPL became aware that the atmospheric steam dump valves are undersized and that the plant cannot be cooled at the rate specified by the FSAR when the offsite power is not available and all the steam, generated by the decay heat, has to be released to the atmosphere. This fact has been reported to the NRC (Reference 5) and a corrective action, consisting of modifying the atmospheric dump valve intervals, was made. FPL found, however, that in order to meet the FSAR criteria, further action was necessary. This action consists of increasing the temperature at which the cooldown of the primary coolant by the shutdown cooling system could be initiated. In justifying the proposed change, FPL has indicated that the shutdown system in the St. Lucie plant is designed to operate at temperatures of up to 350°F (Reference 6), and the change of startup temperature limit from 300°F to 325°F could be accomplished without violating the design criteria. The licensee has also indicated that he has reviewed the FSAR accident analyses and that the proposed change would not adversely affect offsite releases. We have reviewed the licensee's assertion and find that the offsite releases remain unaffected by the change.

We have reviewed FPL's assertions concerning the performance of cooldown systems with the changed temperature limit. We have determined that the increase of this limiting temperature by 25°F will not significantly affect the performance of the shutdown pumps, shutdown heat exchangers or any other components vital to the safe operation of the shutdown cooling system under normal or accident conditions. We conclude therefore that the change of upper temperature limit for the initiation of the shutdown cooling system in the St. Lucie Unit No. 1 plant from 300°F to 325°F will not degrade the performance of the system.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: December 14, 1978

References

1. Florida Power and Light Company letter L-78-67 to NRC (V. Stello), "Proposed Amendment to Facility Operating License DPR-67", dated February 24, 1978.
2. Florida Power and Light Company FSAR, St. Lucie Plant Unit 1 (Docket No. 50-335), Section 6·3·2·2·1.
3. Florida Power and Light Company letter PRN-LI-77-328 to NRC (J. P. O'Reilly), "Reportable Occurrence 335-77-39, St. Lucie Unit 1; Refueling Water Tank Piping", dated October 24, 1977.
4. Florida Power and Light Company letter L-78-70 (R. E. Uhrig) to NRC (V. Stello), "Proposed Amendment to Facility Operating License DPR-67", dated February 27, 1978.
5. Florida Power and Light Company, Reportable Occurrence Report 335-77-22, St. Lucie Unit 1, dated April 22, 1977.
6. Florida Power and Light Company FSAR, St. Lucie Plant, Unit 1 (Docket No. 50-335), Section 9·3·5·2·2.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-335

FLORIDA POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 28 to Facility Operating License No. DPR-67 issued to Florida Power & Light Company (the licensee), which revised Technical Specifications for operation of St. Lucie Plant, Unit No. 1 (the facility), located in St. Lucie County, Florida. The amendment is effective as of its date of issuance.

This amendment revises the Technical Specifications to increase the minimum required volume of water in the Refueling Water Tank and to allow an increase in the coolant temperature at which shutdown cooling is initiated.

The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR Section 51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated February 24 and 27, 1978, (2) Amendment No. 28 to License No. DPR-67, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W. Washington, D. C., and at the Indian River Junior College Library, 3209 Virginia Avenue, Ft. Pierce, Florida. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 14th day of December 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Gerald B. Zwetzig

Gerald B. Zwetzig, Acting Chief
Operating Reactors Branch #4
Division of Operating Reactors