DCS MS-016

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Docket No. 50-335

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Dr. Robert E. Uhrig
Vice President
Advanced Systems & Technology
Florida Power & Light Company
P. O. Box 529100
Juno Beach, Florida 33408

Dear Dr. Uhrig:

The Commission has issued the enclosed Amendment No. 60 to Facility Operating License No. DPR-67 for the St. Lucie Plant, Unit No. 1. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated April 13, 1978, as supplemented.

The amendment revises the Technical Specifications to (1) incorporate limits and surveillance requirements associated with the overpressure mitigation system by the addition of new specifications that define the low temperature reactor coolant system overpressure protection range, (2) incorporate a limit on the maximum primary-to-secondary differential temperature that is permitted prior to starting a reactor coolant pump, (3) incorporate new requirements on the operability of power operated relief valves, (4) add a note to limit the establishment of a high pressure safety injection pump flow path under certain conditions, and (5) revise requirements on the positioning of certain safety injection valves.

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

Sincerely,

Original signed by

Donald E. Sells, Project Manager
Operating Reactors Branch #3
Division of Licensing

Check for Deletions or Comments immediately before issuing

If any come back to EAD

No legal objections

Enclosures:
1. Amendment No. 60 to DPR-67
2. Safety Evaluation

cc w/enclosures:
See next page

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SURNAME	PMKreutzer	DSells/pn	JPMiller	GCLainas	W.D Paton		
DATE	8/1/83	8/7/83	9/2/83	9/6/83	9/9/83		

NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

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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. 60
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 April 13, 1983 as supplemented, 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. 60, 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

James R Miller

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 16, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 60

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.

Pages

Ia
V
X
1-6b (new)
3/4 1-8
3/4 1-12
3/4 4-59 (new)
3/4 4-60 (new)
3/4 5-7
B 3/4 4-15(new)

DEFINITIONS

SECTION | PAGE

1.0 DEFINITIONS

INDEX

DEFINITIONS

SECTION PAGE

INDEX

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

SECTION PAGE

3/4.4 REACTOR COOLANT SYSTEM

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.6 CONTAINMENT SYSTEMS

INDEX

BASES

LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE

1.39 The LOW TEMPERATURE RCS OVERPRESSURE PROTECTIVE RANGE is that operating condition when (1) the cold leg temperature is $\leq$ 275°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.

REACTIVITY CONTROL SYSTEMS

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 makeup tanks is used.

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*When the RCS temperature is less than 165°F, the flow path from the RWT to the RCS via the HPSI pumps shall only be established if the reactor coolant system pressure boundary integrity does not exist, or if no charging pump is operable.

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.

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.

*When the RCS temperature is less than 165°F, the flow path from the RWT to the RCS via the HPSI pumps shall be established only if the reactor coolant system pressure boundary integrity does not exist, or if no charging pump is operable.

REACTOR COOLANT SYSTEM

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.

REACTOR COOLANT SYSTEM

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.

APPLICABILITY: MODES 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.

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 275°F.

*PORVs are not required at Reactor Coolant System temperatures below 165°F when all HPSI pumps and respective injection or header isolation valves are disabled and if a pressurizer bubble is formed with a pressurizer liquid level less than or equal to 40%. PORVs are also 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 45°F reactor coolant pump(s) shall not be started unless the pressurizer liquid level is less than 40%.

APPLICABILITY: MODES 4# and 5.

ACTION:

If a reactor coolant pump is started when the steam generator temperature exceeds primary temperature by more than 45°F and the pressurizer liquid level exceeds 40%, evaluate the subsequent transient to determine compliance with Specification 3.4.9.1.

SURVEILLANCE REQUIREMENTS

4.4.14 Prior to starting a reactor coolant pump, verify either that the steam generator temperature does not exceed primary temperature by more than 45°F or that a pressurizer bubble is drawn and the pressurizer level is equal to or less than 40%.

#Reactor Coolant System Cold Leg Temperature is less than 275°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 215°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 165°F all high pressure safety injection pumps will be disabled and their associated header stop valves closed.

APPLICABILITY: MODES 3*, 4#, and 5.

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 215°F and with more than the allowed high pressure safety injection pumps 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 below 275°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.

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

The low temperature reactor coolant system overpressure mitigating system is provided to prevent RCS overpressurization above the 10 CFR 50, Appendix G, operating limit curves (Figure 3.4-2b or 3.4-2c, as applicable) at RCS temperatures below 275°F. The RCS overpressurization system is based on the use of the pressurizer power operated relief valves (I-V-1402 and I-V-1404) for the design basis mass injection transient, and the formation of a 60% pressurizer bubble by volume for the design basis energy addition transient. For the case when no pressurizer steam bubble is formed, protection against the design basis energy addition transient is derived by limiting the secondary-to-primary temperature differential below 50°F. The operability of the RCS overpressurization protection system will only be required during periods of heatup and cooldown below RCS temperatures below 275°F and periods of cold shutdown when the RCS has pressure boundary integrity.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 60 TO FACILITY OPERATING LICENSE NO. DPR-67

FLORIDA POWER AND LIGHT COMPANY

ST. LUCIE UNIT 1

DOCKET NO. 50-335

I. INTRODUCTION

Incidents identified as "pressure transients" have occurred in pressurized water reactors where the pressure limit in the Technical Specifications for a given temperature is exceeded. These incidents generally occur at relatively low temperatures where the reactor vessel material toughness, i.e., resistance to brittle fracture, is reduced from that which exists at normal operating temperature and where the primary system is completely filled with water, i.e., in a "water-solid" condition.

The "Technical Report on Reactor Vessel Pressure Transients" in NUREG-0138 (Ref. 1) summarizes the technical considerations relevant to this matter, discusses the safety concerns and existing safety margins of operating reactors, and describes the regulatory actions taken to resolve this issue by reducing the likelihood of future pressure transient events at operating reactors.

By letter to the Florida Power and Light Company (FPL) dated August 13, 1976 (Ref. 2), the U.S. Nuclear Regulatory Commission (NRC) requested an evaluation of St. Lucie, Unit 1 to determine susceptibility to overpressurization events and an analysis of these possible events, and required FPL to propose interim and permanent modifications to the systems and procedures to reduce the likelihood and consequences of such events.

By letters dated September 3, 1976 and October 18, 1976, (Refs. 3 and 4), FPL submitted the interim measures that they had taken to minimize the probability of a low temperature-overpressure transient at St. Lucie, Unit 1. FPL participated as a member of a Combustion Engineering (CE) utility task group to determine the long term solution. A generic analysis for CE plants along with possible mitigating systems was provided by a letter dated December 3, 1976 (Ref. 5). FPL submitted additional information regarding the low-temperature overpressure problem including responses to staff concerns and questions, and information about operator awareness and training (Refs. 6, 7, 8, and 9). In an August 23, 1977 letter (Ref. 10), FLP proposed an interim overpressure mitigating system (OMS) using the existing pressurizer power operated relief valves (PORVs) with a variable low pressure setpoint. They later proposed that their interim system be considered as the final OMS (Ref. 11), and in a letter dated April 13, 1978 (Ref. 12), the system was described, an analysis was provided, and technical specifications were proposed for the OMS. However, as stated in Reference 15, the St. Lucie 1 OMS no longer has the capability to automatically vary the pressure setpoint as reactor coolant temperature changes. The automatic setpoint variation feature has been permanently removed.

This evaluation is of only the physical and operational aspects of the St. Lucie Unit 1 low temperature overpressure mitigating system. The evaluation of the electrical, instrumentation, and control aspects is not included.

II. REVIEW CRITERIA

The NRC formally addressed reactor vessel overpressurization in August 1976, and requested that the utilities provide a solution to the problem. The design criteria were subsequently identified through meetings and correspondence with utility representatives. NUREG-0224, "Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors" (Ref. 13), with appended Branch Technical Position RSB 5-2 (Ref. 14) states the staff's requirements for the overpressure protection system.

III. DESCRIPTION

The St. Lucie, Unit 1 OMS consists of two separate trains, each containing a power-operated relief valve (PORV), an isolation valve and associated circuitry. When in the low pressure mode the system provides a low pressure setpoint for both PORV trains. When the system is enabled, it will terminate all analyzed pressure transients below the Technical Specification limit (0 to 40 Effective Full Power Years, EFPY) by automatically opening the PORVs. A three-position PORV mode selector switch is used to enable and disable the low setpoint of each relief valve. An enabling alarm which monitors system temperature and pressure is provided to alert the control room operator to enable the overpressure mitigating system when either the reactor cooling system (RCS) temperature falls to 275°F or RCS pressure decreases to 400 psig. This alarm will not clear unless the PORV mode selector switch is in the low setpoint position and the MOVs upstream of the PORVs indicate "open". If the reactor coolant pressure comes within 25 psi of the low pressure setpoint, an alarm will alert the operator of pending PORV actuation. Should the reactor coolant pressure reach the low pressure setpoint, an alarm will inform the operator that the PORVs have received a signal to open. When both RCS pressure and temperature exceed specified setpoints, an alarm alerts the operator to return the PORVs to their normal setpoint. At a reactor coolant temperature of 275°F, a high temperature interlock removes the OMS from service. This interlock provides assurance that the OMS will not be inadvertently actuated at power. This RCS temperature is the highest temperature anticipated for OMS operation. (The isothermal pressure limit at 275°F is approximately 1800 psia, which is well above the shutoff head of the high pressure safety injection (HPSI) pumps.)

The St. Lucie pressurizer PORVs are pilot-operated self-actuating valves. The pilot mechanism is comprised of a solenoid operated pilot valve that vents the pressure from one side of an operating piston allowing system pressure to actuate the relief valve. The orifice area of each PORV is 1.354 square inches and the opening times are approximately 3 milliseconds. A single PORV has sufficient capacity to mitigate all of the postulated overpressure events except for an inadvertent HPSI System actuation and for a reactor coolant pump (RCP) start with a 150°F ΔT between the RCS and a steam generator. For these events administrative controls have been instituted to restrict conditions so that a single PORV can mitigate the overpressure event.

IV. EVALUATION

A. System

1. Testability

The staff position requires that a test be performed to assure operability of the system electronics prior to each shutdown and that a test for valve operability, as a minimum, be conducted as specified in the ASME Code Section XI. FPL performs channel calibration and full stroke testing of the solenoid-operated pilot valve once per 18 months and has proposed procedures to verify operability prior to each shutdown. These testing requirements shall be incorporated into the St. Lucie-1 Technical Specifications.

The staff concludes that, with this addition to the Technical Specifications, the testing requirement will be met.

2. Single Failure Criteria

The specified single failure criterion for the overpressure mitigating system is that it should be designed to protect the vessel given a single failure in addition to the failure that initiated the pressure transient. The St. Lucie, Unit 1 OMS meets this criteria for all cases reviewed except for the case where the initiating event is a loss of power from one DC control bus when a charging pump, that is powered from another DC bus, is running and the RCS is in a "water solid" condition. A loss of power then would result in isolation of the letdown line and make one PORV inoperable. Because the other PORV is powered from the other DC control bus, it will remain functional. However, when the single failure is postulated for the remaining PORV, no low-temperature overpressure protection would be available.

In order to comply with the single failure criteria for this event, FPL by Reference 15 has committed to operating procedures which shall minimize the time the RCS is in the "water solid" condition and which shall restrict maintenance and testing activities involving DC buses so they do not take place while the RCS is in a "water solid" condition. Since maintenance and testing errors are significant contributors to DC bus failures, we find that these changes in operating procedures will make it highly unlikely that this single failure event will occur during the life of St. Lucie Unit 1.

With these changes in operating procedures, the staff finds that the St. Lucie Unit 1 OMS meets the single failure criteria.

3. Seismic Design

The specified seismic criteria is that the Overpressure Mitigating System should be designed to function during an Operating Basis Earthquake (OBE).

The St. Lucie Unit 1 electromatic PORVs were designed and manufactured in accordance with ASME Boiler and Pressure Vessel Code Section III and are Class I valves.

The staff concludes that the St. Lucie-1 Overpressure Mitigation System meets the seismic criteria.

B. Analysis

The analyses are divided into two general categories of pressure transients: mass input from sources such as charging pumps, safety injection pumps, and accumulator tanks; and energy input, which causes thermal expansion, from sources such as steam generators and decay heat.

FPL conducted the analyses by first determining the worst case overpressurization events for both of these general categories, and then evaluating the effectiveness of the OMS in terminating these worst case events. All analyses were performed assuming water-solid RCS conditions. This assumption is conservative since it eliminates any time delay in transient response due to a vapor space in the pressurizer. Also, all letdown flow paths that could mitigate a particular overpressurization event were considered isolated.

1. Mass Input Cases

The analysis submitted for St. Lucie, Unit 1 by FPL considered the following overpressurization mass input events:

1. Inadvertent Safety Injection (SI) actuation

2. Inadvertent start of a single High Pressure Safety Injection (HPSI) pump

3. Inadvertent mismatch of charging and letdown flow.

The Low Pressure Safety Injection (LPSI) pumps and the SI accumulator tanks were not considered as contributing mass inputs since the LPSI pump's shut-off head (180 psi) and the SI accumulator tank design pressure (250 psig) are below the Technical Specification limits.

The analyzed inadvertent SI events included actuation of two HPSI pumps with all three charging pumps, actuation of a single HPSI pump with three charging pumps, and actuation of only the three charging pumps when all HPSI pumps are disabled.

The results of the analyses show that inadvertent Safety Injection (SI) actuation (the case with two HPSI and three charging pumps running) is the most severe mass input event. This event was evaluated further assuming one PORV and two PORVs available in the low setpoint mode. With one PORV available, the RCS would reach an equilibrium pressure of 800 psia which corresponds to a Technical Specification temperature of 195°F. With two PORVs available, the equilibrium pressure would be 470 psia which corresponds to a Technical Specification temperature of 95°F. Once a PORV opens, it will remain open as long as the equilibrium pressure is above the valve blowdown closure setting. If the equilibrium pressure becomes less than the blowdown setpoint, the peak RCS pressure will equal the valve set pressure and valve cycling will occur. For equilibrium pressures above the closure setpoint, the valve or valves will remain open until operator action secures the input flow.

The results of these analyses show that the PORVs do not have sufficient capacity to mitigate the most severe mass input event for all RCS temperatures below the minimum pressurization temperature. FPL has therefore instituted operating instructions and proposed administrative controls to provide a reasonable degree of assurance that the Technical Specification limits will not be exceeded during an inadvertent SI actuation. As stated above, a single PORV will mitigate the resultant overpressure transient for all RCS temperatures above 195°F. Proposed St. Lucie, Unit 1 technical specifications state that prior to decreasing RCS temperature below 215°F a maximum of one HPSI pump is to be operable. With one HPSI pump and three charging pumps in operation, one PORV will mitigate an inadvertent SI transient at 550 psia which corresponds to 155°F on the MPT curve. The proposed technical specifications state that prior to decreasing the RCS temperature below 165°F, all HPSI pumps must be disabled with their associated header stop valves closed. A single PORV has sufficient capacity to relieve the volume of coolant supplied by three charging pumps or a single HPSI pump by itself. Therefore, an inadvertent SI actuation that results in either of the above situations, would be mitigated with a single PORV cycling at its setpoint until operator action is taken to stop the mass addition. For all temperatures below 165°F the PORV setpoint is 465 psia which corresponds to a minimum allowable isothermal temperature of 95°F.

The staff concludes that, when the proposed administrative controls are incorporated into the Technical Specifications, the St. Lucie, Unit 1 OMS will satisfactorily mitigate all mass addition events.

2. <u>Energy Input Cases</u>

The St. Lucie, Unit 1 analysis considered the following energy input overpressurization events:

1. Decay heat addition due to shutdown cooling system isolation

2. Inadvertent pressurizer heater input

3. Energy input from the steam generator secondary to the primary coolant subsequent to operation of a reactor coolant pump (RCP) when the steam generators are at a higher temperature than the reactor vessel inventory.

Energy addition analyses determined the RCS pressure response as a function of time. After each time increment the RCS pressure was determined as a function of the average liquid system enthalpy and average liquid specific volume. The system enthalpy changes according to the heat addition rate. For analyses that assume no liquid relief capability, the specific volume of the system is considered a constant since pressure boundaries are assumed fixed and system mass remains constant. Other conservative assumptions include isolated letdown and no sensible heat absorption by the RCS component metal mass.

From the analysis it was determined that the Reactor Coolant Pump (RCP) start transient is the most severe of the energy addition events analyzed. The other two energy addition events result in a rate of pressure increase that is relatively slow in comparison to the rate of pressure increase indicated for the RCP start transient. These two events are bounded by the three charging pump mass addition event and can be adequately mitigated by a single PORV.

A RCP operating when the steam generators are at a higher temperature than the reactor vessel will cause a rapid pressure increase in a water-solid RCS. The energy addition rate is not constant and so a computer model that represents the RCS by five nodes was employed to simulate the resulting water-solid RCS pressure response. Assuming an instantaneous RCP start, no heat absorption or metal expansion at the primary pressure boundaries, full pressurizer heater input, and one-percent decay heat as a conservative upper bound, RCS pressure was computed as a function of time.

With both PORVs functioning properly, the maximum allowable ΔT between the RCS and the steam generator is approximately 150°F. However, to conform with the single failure criteria, only one PORV is considered in the analysis. With only one PORV, a ΔT of 50°F between the RCS and the steam generators was used.

The proposed technical specifications state that a reactor coolant pump shall not be started if the ΔT is more than 45°F unless the pressurizer liquid level is less than 40%. With this large of a steam bubble, a RCP start transient would be mitigated before it reached the low pressure PORV setpoint.

The analysis was performed using an initial RCS pressure of 300 psia, a 50°F ΔT, and RCS temperature of 150°F. The PORV opens at the set pressure of 465 psia and limits the peak RCS pressure to 490 psia which corresponds to a minimum RCS temperature of 105°F based on the isothermal (heatup) pressure-temperature limit.

The results of this analysis show that when using a 50°F ΔT limitation, the OMS, with only one PORV functioning, provides assurance that the Technical Specification limits will not be exceeded for the 465 psia setpoint. Thus in all cases, for the given RCS temperature, the Technical Specification limits are not exceeded; therefore, the performance of the St. Lucie 1 OMS is judged to be adequate for heat induced transients.

C. Administrative Controls

A number of provisions for the prevention of pressure transients have been incorporated in the plant operating procedures. Some examples of these are:

1. Heatup and normal cooldown procedures minimize the time the plant is in a water solid condition.

2. During plant heatup, a steam bubble is drawn in the pressurizer before the Shutdown Cooling System is removed from service.

3. During plant cooldown, the Shutdown Cooling System is placed in service prior to collapsing the pressurizer steam bubble.

4. Procedures have been revised to eliminate initiation of charging pump flow without adequate letdown capability during operations.

5. The procedure for fill and vent of the RCS has been modified so that the second backup charging pump control switch is placed in the "off" position to preclude auto start of the pump.

6. The procedure for testing the HPSI pumps during shutdown was changed to require closing the motor-operated pump discharge valves.

The following procedures shall be required by Technical Specifications:

1. If the steam generator temperature exceeds the primary temperature by more than 45°F a reactor coolant pump shall not be started unless the pressurizer liquid level is less than 40%.

2. When Reactor Coolant System cold leg temperature is below 275°F, two power operated relief valves (PORVs) shall be operable, with their setpoints selected to the low temperature modes of operation.

3. Before decreasing the reactor coolant system temperature below 215°F, a maximum of only one high pressure safety injection pump is to be operable with its associated header stop valves open.

4. Prior to decreasing the Reactor Coolant System temperature below 165°F all high pressure safety injection pumps shall be disabled and their associated header stop valves closed. The high pressure safety injection pumps shall be verified inoperable and the associated header stop valves closed prior to decreasing below the above specified RCS temperature and once per month when the RCS is at refueling temperatures.

With the incorporation of these 4 items into the Technical Specifications these administrative controls will be acceptable.

V. CONCLUSIONS

The administrative controls and plant modifications proposed by the Florida Power and Light Company provide protection for St. Lucie, Unit 1 from pressure transients at low temperatures by reducing the probability of initiation of a transient and by limiting the pressure of such a transient to below the limits set by 10 CFR Part 50 Appendix G. The staff finds that, with the addition of Technical Specification as stated above, the St. Lucie-1 overpressure mitigating system meets GDC 15 and 31 and that FPL has implemented the guidelines of NUREG-0224. The St. Lucie, Unit 1 overpressure mitigating system is judged to be an adequate solution to the problem of transients at low pressure and temperature.

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 appriasal need not be prepared in connection with the issuance of this amendment.

Conclusion

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

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3. FPL letter (Uhrig) to NRC (Ziemann), dated September 3, 1976.

4. FPL letter (Uhrig) to NRC (Ziemann), dated October 18, 1976.

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6. FPL letter (Uhrig) to NRC (Ziemann), dated February 28, 1977.

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13. Zech, G.; Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors; U.S. NRC NUREG-0224; September, 1978.

14. U.S. NRC; Standard Review Plan; NUREG-0800; pages 5.2.2-7 & 5.2.2-8; July, 1981.

15. FPL letter (Uhrig) to NRC (Clark), dated April 6, 1983.