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Docket No. 50-302

Mr. W. P. Stewart
Director, Power Production
Florida Power Corporation
Post Office Box 14042, Mail Stop C-4
St. Petersburg, Florida 33733

Dear Mr. Stewart:

The Commission has issued the enclosed Amendment No. 21 to Facility Operating License No. DPR-72 for Crystal River Unit No. 3 Nuclear Generating Plant. This amendment consists of changes to the license and its appended Technical Specifications in response to portions of your applications dated January 23, 1978, as supplemented July 3, 1979.

This amendment revises the Technical Specifications to allow "racking out" the power supply breakers for the high pressure injection isolation valves during Mode 4 operation. These flow paths are isolated as part of the overpressure mitigating system at the facility. Further changes regarding this system will be proposed by the licensee within 30 days from the date of this letter.

This action satisfies the requirements of license condition 2.C.(6). This condition is therefore removed from the license.

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

Sincerely,

Original signed by
Robert W. Reid

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

Enclosures:
1. Amendment No. 21 to DPR-72
2. Safety Evaluation
3. Notice

cc w/encl:
See next page

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DATE	7/3/79	7/3/79	7/3/79	7/3/79	7/3/79	7/3/79

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Director, Power Production
Florida Power Corporation
P. O. Box 14042, Mail Stop C-4
St. Petersburg, Florida 33733

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This amendment revises the Technical Specifications to delete the requirement to maintain an operable high pressure injection flow path during Mode 4 operation. These flow paths are isolated as part of the overpressure mitigating system at the facility. Further changes regarding this system will be proposed by the licensee within 30 days from the date of this letter.

This action satisfies the requirements of license condition 2.C.(6). This condition is therefore removed from the license.

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

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cc w/enclosures: See next page

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SURNAME	RIngram	CNelson/cb	RReid	WGammill	S H Lewis	S Weiss
DATE	7/3/79	7/3/79	7/3/79	7/3/79	7/3/79	7/3/79

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

July 3, 1979

Docket No. 50-302

Mr. W. P. Stewart
Director, Power Production
Florida Power Corporation
Post Office Box 14042, Mail Stop C-4
St. Petersburg, Florida 33733

Dear Mr. Stewart:

The Commission has issued the enclosed Amendment No. 21 to Facility Operating License No. DPR-72 for Crystal River Unit No. 3 Nuclear Generating Plant. This amendment consists of changes to the license and its appended Technical Specifications in response to portions of your applications dated January 23, 1978, as supplemented July 3, 1979.

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Robert W. Reid

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

Enclosures:
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2. Safety Evaluation
3. Notice

cc w/encl:
See next page

Florida Power Corporation

cc w/enclosure(s):
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Vice President and General Counsel
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St. Petersburg, Florida 33733

Mr. Wilbur Langely, Chairman
Board of County Commissioners
Citrus County
Iverness, Florida 36250

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Atlanta, Georgia 30308

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Bureau of Intergovernmental Relations
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Tallahassee, Florida 32304

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

FLORIDA POWER CORPORATION
CITY OF ALACHUA
CITY OF BUSHNELL
CITY OF GAINESVILLE
CITY OF KISSIMMEE
CITY OF LEESBURG
CITY OF NEW SMYRNA BEACH AND UTILITIES COMMISSION, CITY OF NEW SMYRNA BEACH
CITY OF OCALA
ORLANDO UTILITIES COMMISSION AND CITY OF ORLANDO
SEBRING UTILITIES COMMISSION
SEMINOLE ELECTRIC COOPERATIVE, INC.
CITY OF TALLAHASSEE

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 21
License No. DPR-72

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

   A. The application for amendment by Florida Power Corporation, et al (the licensees) dated January 23, 1978, as supplemented July 3, 1979, 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.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraphs 2.C.(2) and 2.C.(6) of Facility Operating License No. DPR-72 are hereby amended as follows:

   A. Revise paragraph 2.C.(2) in its entirety to read:

      Technical Specifications

      The Technical Specifications contained in Appendices A and B, as revised through Amendment No.21 , are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

   B. Delete paragraph 2.C.(6) in its entirety.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 3, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 21

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

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

3/4 5-6

ECCS SUBSYSTEMS - T_{avg} < 280°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 injection (HPI) pump,

b. One OPERABLE low pressure injection (LPI) pump,

c. One OPERABLE decay heat cooler, and

d. An OPERABLE flow path[1/] capable of taking suction from the borated water storage tank (BWST) and transferring suction to the containment emergency sump.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the HPI pump or the flow path from the borated water storage tank, restore at least one ECCS subsystem to OPERABLE status within one hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the decay heat cooler or LPI pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 280°F by use of alternate heat removal methods.

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

1/ The high pressure injection isolation valves may have their power supply breakers "racked out" in Mode 4.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 21 TO LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

1.0 Introduction

By letter dated December 2, 1976 (Reference 1), Florida Power Corporation (the licensee or FPC) submitted to the NRC a plant-specific analysis in support of the reactor vessel overpressure mitigating system (OMS) for Crystal River Unit 3 Nuclear Generating Plant (CR-3). The analysis was supplemented by letter dated February 17, 1979 (Reference 2) and other documentation submitted by FPC (References 3, 4, 5). FPC has installed the equipment and incorporated the procedures described in this report. Hence, this report summarizes past efforts by the licensee, vendor, and NRC staff.

Currently license condition 2.C.(6) of the operating license for CR-3 requires the installation of a long-term means of protection against reactor coolant system overpressurization prior to restart from the current outage.

NRC staff review of all information submitted by FPC in support of the proposed overpressure mitigating system is complete and has found that the system provides adequate protection from overpressure transients. A detailed safety evaluation follows.

2.0 Background

Over the last few years, incidents identified as pressure transients have occurred in pressurized water reactors (PWRs). This term "pressure transients," as used in this report, refers to events during which the temperature pressure limits of the reactor vessel, as shown in the facility Technical Specifications, are exceeded. All of these incidents occurred at relatively low temperature (less than 200°F) where the reactor vessel material toughness (resistance to brittle failure) is reduced.

The "Technical Report on Reactor Vessel Pressure Transients" in NUREG 0138 (Reference 6) 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. A brief discussion is presented here.

2.1 Vessel Characteristics

Reactor vessels are constructed of high quality steel made to rigid specifications, and fabricated and inspected in accordance with the time-proven rules of the ASME Boiler and Pressure Vessel Code. Steels used are particularly tough at reactor operating conditions. However, since reactor vessel steels are less tough and could possibly fail in a brittle manner if subjected to high pressures at low temperatures, power reactors have always operated with restrictions on the pressure allowed during startup and shutdown operations.

At operating temperatures, the pressure allowed by Appendix G limits is in excess of the setpoint of currently installed pressurizer code safety valves. However, most operating PWRs did not have pressure relief devices to prevent pressure transients during cold conditions from exceeding the Appendix G limit.

2.2 Regulatory Actions

By letter dated October 1, 1976 (Reference 7), the NRC requested that FPC begin efforts to design and install plant systems to mitigate the consequences of pressure transients at low temperatures. It was also requested that operating procedures be examined and administrative changes be made to guard against initiating overpressure events. It was felt by the NRC staff that proper administrative controls were required to assure safe operation for the period of time prior to installation of the proposed overpressure mitigating hardware.

FPC responded (Reference 1) with information describing measures to prevent these transients along with some discussion of proposed hardware. The proposed hardware change was to install a low pressure actuation setpoint on the existing pressurizer pilot operated relief valves (PORVs).

Additional NRC staff concerns were expressed in letters to FPC dated November 19, 1976, January 7, 1977, and November 11, 1977 (References 8, 9, 10). FPC responded to these concerns in References 2 and 4. The correspondence focused on system design criteria discussed below.

2.3.1 Design Criteria

Through this series of meetings and correspondence with PWR vendors and licensees, we developed a set of criteria for an acceptable overpressure mitigating system. The basic criterion is that the mitigating system will prevent reactor vessel pressures in excess of those allowed by Appendix G. Specific criteria for system performance are:

1) Operator Action: No credit can be taken for operator action for ten minutes after the operator is aware of a transient.

2) Single Failure: The system must be designed to relieve the pressure transient given a single failure in addition to the failure that initiated the pressure transient.

3) Testability: The system must be testable on a periodic basis consistent with the system's employment.

4) Seismic and IEEE 279 Criteria: Ideally, the system should meet seismic Category I and IEEE 279 criteria. The basic objective is that the system should not be vulnerable to a common failure that would both initiate a pressure transient and disable the overpressure mitigating system. Such events as loss of instrument air and loss of offsite power must be considered.

We also instructed the licensee to provide an alarm which monitors the position of the pressurizer relief valve isolation valves, along with the low setpoint enabling switch, to assure that the overpressure mitigating system is properly aligned for shutdown conditions.

Licensees were informed that their proposed mitigating systems were to meet these criteria for the most adverse of hypothesized scenarios, that is, the largest mass or heat addition which could occur at the specific plant. While administrative procedures were to be employed to reduce the probability of an initiating event, administrative procedures were not to be employed in lieu of hardware modifications. These hardware modifications were to provide sufficient relief capacity to mitigate the most adverse scenario.

It was recognized that these criteria were of a general nature and that exceptions would be required as individual reviews progressed. (See Section 3.1 Evaluation.)

2.4 Design Basis Events

The incidents that have occurred to date have been the result of operator errors or equipment failures. Two varieties of pressure transients can be identified: a mass input type from charging pumps, safety injection pumps, safety injection accumulators; and a heat addition type which causes thermal expansion from sources such as steam generators or decay heat.

Only one overpressure event at low temperature (during hydrostatic test) has occurred at a Babcock and Wilcox (B&W) nuclear supplied steam system (NSSS). The most common cause of overpressure transients to date has been isolation of the letdown path. We have identified the most limiting mass input transient to be inadvertent injection by the largest safety injection pump. The most limiting thermal expansion transient is the start of a reactor coolant pump with a large temperature difference between the water in the reactor vessel and the water in the steam generator.

FPC has provided an evaluation of:

a. Erroneous actuation of the High Pressure Injection (HPI) system.

b. Erroneous opening of the core flood tank discharge valve.

c. Erroneous addition of nitrogen to the pressurizer.

d. Makeup control valve (makeup to the Reactor Coolant System (RCS)) fails full open

e. All pressurizer heaters erroneously energized.

f. Temporary loss of the Decay Heat Removal (DHR) System's capability to remove decay heat from the RCS.

g. Thermal expansion of RCS after starting a reactor coolant pump (RCP) due to stored thermal energy in the steam generator.

3.0 System Description

The OMS consists of active and passive subsystems. The active subsystem is simply the modification of the actuation circuitry of the existing electrical PORV to provide dual setpoints, a normal operation setpoint of 2450 psig and a low pressure setpoint of 550 psig. The low pressure setpoint is employed when the RCS is below 280°F. This system is manually enabled. An alarm will function should the operator fail to enable the system. An alarm has also been installed to monitor the position of the pressurizer relief block valve, RC-V2. The passive subsystem consists of the introduction of a nitrogen blanket at the top of the pressurizer. The reactor is operated during heatup and cooldown with a steam or nitrogen bubble. The bubble functions as a mechanical damper. This subsystem is part of the original B&W design.

3.1 System Evaluation

The CR-3 OMS is both redundant and functionally diverse. The plant, by virtue of a gas (nitrogen or steam) blanket in the pressurizer and the relatively small size, and hence heat capacity, of the once through steam generators, is not susceptible to heat addition transients. The plant is never operated in a water solid condition.

In contrast, the OMS of a Westinghouse or Combustion Engineering NSSS consists of two relief valves with independent low setpoint actuation circuitry. The two trains are identical, i.e., not diverse. (It is noted the diversity although desirable was never an NRC staff design criteria.) These systems are susceptible to heat addition transients. These systems are operated in a water solid condition.

FPC has submitted analyses of the design bases events shown in Section 2.4 (Reference 2). We accept these analyses.
These analyses show that, in the event of a postulated mass addition, actuation of the relief valve will limit RCS pressures to the relief valve setpoint and hence below Appendix G limits. Should the relief valve fail closed, or actuation circuitry fail, the system pressure would continue to increase. With the exception of postulated high pressure safety injection, the nitrogen bubble in the pressurizer will provide at least ten minutes, and in some cases substantially longer time, for operator action. The analyses also show that in the event that decay heat removal was lost, more than 29 minutes would pass before the relief valve setpoint would be reached. Postulated RCP starts with steam generator secondary water temperature greater than primary water temperature will not result in RCS pressure increases to the relief valve setpoint value. Hence, CR-3 is not considered susceptible to overpressure transients due to inadvertent heat addition.

System pressure overshoot, that is, increase of primary coolant pressure after pressure reaches the low setpoint value, does not occur on B&W NSSS due to the rapid action of the electrical PORV and the relatively slow rates of pressure increase due to the nitrogen blanket in the pressurizer.

The CR-3 OMS is tolerant of seismic events. FPC has performed analyses for the pilot assembly connection pipe assuming seismic motion of 3.0 horizontal and 3.0 vertical. The actual valve meets Class 1 requirements. Testing with simulated seismic loadings has not been performed. This was not a requirement at the time the plant was designed and constructed. Even if it is assumed that the valve, connection pipe, or actuation circuitry, failed due to a seismic event, the nitrogen blanket in the pressurizer would provide protection for postulated low temperature overpressure events.

The system is testable and is to be tested prior to use. The PORV is to be tested each shutdown.

The system does not strictly meet 1EEE279 criteria. The basic objective of this criterion, prevention of common mode failure, is met by virtue of the subsystem diversity.

For all postulated heat addition transients and for all mass additions other than inadvertent high pressure safety injection, the CR-3 OMS meets single failure and operator action criteria.

In the event that the largest possible mass addition were to occur, one HPI train, actuation of the relief valve would terminate the transient. Should this valve fail, the RCS pressure would exceed system pressure in four to five minutes (depending on the initial system conditions). Hence, for this postulated event, the system does not meet single failure/operator action criteria. For lesser mass addition rates, in the event that the relief valve failed, the pressurizer bubble would act as a pressure damper providing more than ten minutes for operator action.

In contrast, the OMS of a Westinghouse or Combustion Engineering NSSSs will (with specific plant exceptions), assuming that one of the two relief valves or associated circuitry were to fail, terminate this transient.

Administrative controls to mitigate HPI must be found acceptable or additional hardware installed. Both options were considered and are discussed below.

A makeup/charging pump is run to provide RCS main coolant pump seal water. Actuation of a HPI train consists of opening a HPI motor operated valve, MOV, permitting flow from the makeup/charging pump to the RCS. Circuit breakers for the closed HPI MOVs are "racked out" and "tagged" during plant cooldown. With the motor operator "racked out", flow through the valve would represent a passive failure and need not be considered. One must insure that these valves are closed when HPI is not needed without decreasing the probability that they can be opened when HPI is needed.

Inadvertent HP safety injection is of interest during cooldowns from approximately 280°F to 150°F. The licensee has estimated that the time during the six typical cold shutdowns a year the RCS is between these temperatures is approximately one and one-half hours. Above this temperature, the vessel can withstand higher pressures. Below this temperature the RCPs and, hence, the operating makeup/charging pump which supplies seal water to the RCP would be shutdown. In order to initiate an event it would then be necessary to: (1) inadvertently turn on the makeup pump(s), and (2) inadvertently open a HPI valve(s). The pumps and valves are both "racked out."

To preclude HPI in the temperature range of 280°F to 150°F, the licensee must "rack out" the HPI isolation valve circuit breakers with the valves in their normally closed position. HPI pump operation within this range is still necessary for makeup and RCP seal flow.

In order to insure that the operator "racks out" the HPI valves, the licensee has installed an alarm to alert the operator if the RCS temperature is below 280°F and the valves have not been "racked out." With power to the valve motor operators, the valves would open on a safety injection actuation signal. This signal is bypassed during normal depressurization at a pressure of 1750 psig. Failure to follow this procedure will also result in an alarm.

Hardware modifications were discussed with FPC and other B&W NSSS licensees. Metropolitan Edison Company submitted several options and associated costs (Reference 11). While actual plant modifications and costs would vary amongst licensees, it is believed that these options are representative of possible hardware modifications.

Options considered include: modification of the DHR system, modification of the makeup and purification system, addition of a second pressure relief valve on the pressurizer. These options were estimated to cost $200,000 to $400,000. These options introduce additional safety concerns.

Relief capacity addition to the DHR system is only of value with respect to low temperature overpressurization when the DHR is aligned. This system is automatically blocked at an RCS pressure of 284 psig. Modification of the system would require modification of the DHR autoclosure interlocks. Spurious failure of these modified interlocks would increase the probability of primary breaks outside of containment. Installation of relief and block valves downstream of the HPI valves (that is, modification of the makeup system) would increase the probability that HPI, if required, would be impaired.

Hence, although these hardware modifications would comply with the letter of our guidelines, they are not considered necessary. Administrative controls supplemented by the single pressure relief train, and pressure and level indication and alarms are considered a suitable and acceptable alternative.

Credit for administrative controls is consistent with past NRC staff actions. We have permitted a manually enabled system, credit for blocking safety injection actuation signal, credit for successfully blocking one of two high pressure safety injection trains, and credit for blocking accumulator injection. On Combustion Engineering and Westinghouse NSSSs we have assumed administrative control of the primary to secondary differential temperature for heat addition analyses. For B&W NSSS, we have assumed that the nitrogen bubble will be established (a manual procedure) and that the initial pressurizer level will be controlled.

4.0 Administrative Controls

To supplement the hardware modifications and to limit the magnitude of postulated pressure transients to within the bounds of the analysis provided by the licensee, a defense in depth approach is adopted using procedural and administrative controls. Specific conditions required to assure that the plant is operated within the bounds of the analysis are described below.

4.1 Procedures

A number of provisions for prevention of pressure transients are incorporated in the plant operating procedures.

1) The OMS is to be manually enabled when the reactor coolant system temperature is less than 280°F. The low pressure setpoint is 550 psig. An alarm will sound if the operator fails to enable the system. This requirement is to be incorporated in the plant Technical Specifications. An alarm will also be actuated if the operator closes the PORV isolation valve and the RCS temperature is below 280°F.

2) The plant is to be operated with a steam or nitrogen blanket in the pressurizer during plant cooldowns and heatups. The initial pressurizer water level is to be less than or equal to the high level alarm at system pressures above 100 psig and less than the high high level alarm for pressures less than or equal to 100 psig.

3) The makeup tank water level is to be less than the high level alarm.

4) Core Flood Tank discharge valves are to be closed and circuit breakers for the motor operators "racked out" during plant cooldown before the RCS pressure is decreased to 700 psig. The valve positions are also alarmed. This is normal procedure.

5) HPI MOVs are "racked out" during plant cooldown prior to reaching 280°F system temperature. Power to these valves is also alarmed.

Extensive use of alarms insures that the operator is aware of vital plant conditions outside the bounds of those assumed in the safety analysis. The operator must take corrective action to clear these alarms. Overpressurization of the vessel might only occur if an initiating event was coincident with ignoring these alarms.

6) Testing of HPI pumps during shutdown will only be performed with the vessel head removed.

We find that the procedural and administrative controls described are acceptable.

4.2 Technical Specifications

To ensure operation of the low temperature overpressure mitigating system, and decrease the probability that an initiating event which will challenge the system occurs, FPC has proposed (Reference 5) to incorporate operability requirements of the pressurizer relief valve in the plant Technical Specifications. While operability requirements for the relief valve are necessary, we have determined that additional Technical Specifications are also necessary. These specifications would require that plant parameters, such as pressurizer level, are maintained within the limit assumed in the analyses and plant instruments, such as pressurizer level instruments, are operable when relied upon by operators to maintain parameters within specified limits. Therefore, the proposed Technical Specification is not sufficient. By letter dated July 3, 1979, the licensee has committed to propose changes to the Technical Specifications to address the concern discussed above within 30 days.

As noted previously, the licensee will make the HPI isolation valves inoperable for automatic and remote manual operation when below 280°F (Mode 4, 5 and 6). Currently Technical Specification 3.5.3 requires that one HPI pump and flow path be operable during Mode 4. Table 3.3-3 in the Technical Specifications further defines this requirement as the capability to manually initiate HPI during Mode 4.

"Racking out" the power supply to the HPI isolation valves still allows manual initiation of HPI by "racking in" the breakers and then operating the valves. However, this requires operator action outside the control room. We have independently verified that in the unlikely event of a loss of coolant accident which does not depressurize the reactor coolant system such that low pressure injection is functional, the operators have adequate time (greater than 30 minutes) to initiate high pressure injection. This assumes no credit for make-up flow.

Based on the above, we conclude that "racking out" the power supply breakers to the high pressure injection valves in Mode 4 is acceptable. We will add a note to Technical Specification 3.5.3 to clarify this requirement.

5.0 Conclusion

The administrative controls and hardware changes proposed by FPC provide protection for CR-3 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 Appendix G. We find that the overpressure mitigating system is acceptable as a long-term solution to the problem of overpressure transients and satisfies the requirements of license condition 2.C.(6).

During its review the NRC staff identified certain features which, although not necessary for satisfactory operation of the OMS, would be beneficial in the event a pressure transient occurs at low temperature. These additional features would provide direct indication that the transient was in progress and ensure that the transient was recorded for later evaluation. In its letter of June 27, 1979, the licensee proposed to implement the following modifications by the end of the next scheduled refueling outage.

1) A pressure alarm with a setpoint below that of the PORV setpoint to give the operator direct indication of a low temperature pressure transient in progress and that the RCS pressure is on a trend that might exceed the PORV setpoint (550 psig).

2) Recorder(s) to continuously record RCS pressure and temperature. This will provide a permanent record of all low temperature pressure transients. Pressure recorders with a capability in the range of 100 psig per second recording are being investigated.

We find these proposed modifications provide the benefits discussed above and the schedule for implementation is acceptable.

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: July 3, 1979

References:

1) J. T. Rodgers, "Interim Response to Overpressurization at Shutdown Conditions," December 2, 1976, FPC letter.

2) J. T. Rodgers, "Response to NRC Request for Additional Information," February 17, 1977, FPC letter.

3) J. T. Rodgers, Status report to John Stolz, NRC, June 3, 1977, FPC letter.

4) W. P. Stewart, "Response to NRC Request for Additional Information," January 5, 1978, FPC letter.

5) W. P. Stewart, "Technical Specification Change Request No. 17," January 23, 1978, FPC letter.

6) "Staff Discussion of Fifteen Technical Issues Listed in Attachment G, November 3, 1976 Memorandum from Director NRR to NRR Staff," NUREG-0138, November 1976.

7) J. F. Stolz, "Verification for Compliance with Appendix G Pressure-Temperature Limits During Startup and Shutdown," NRC letter, October 1, 1976.

8) J. F. Stolz, NRC letter to FPC in regard to overpressure protection system, NRC letter, November 19, 1976.

9) J. F. Stolz, "Verification for Compliance with Appendix G Pressure-Temperature Limits During Startup and Shutdown," (Crystal River Unit 3 Nuclear Generating Plant), NRC letter, January 7, 1977.

10) R. W. Reid, NRC letter to FPC in regard to overpressure protection system, November 11, 1977.

11) J. G. Herbein, "Overpressure Protection System," January 13, 1978, Met-Ed letter GQL 0048.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-302

FLORIDA POWER CORPORATION
CITY OF ALACHUA
CITY OF BUSHNELL
CITY OF GAINESVILLE
CITY OF KISSIMMEE
CITY OF LEESBURG
CITY OF NEW SMYRNA BEACH AND UTILITIES COMMISSION, CITY OF NEW SMYRNA BEACH
CITY OF OCALA
ORLANDO UTILITIES COMMISSION AND CITY OF ORLANDO
SEBRING UTILITIES COMMISSION
SEMINOLE ELECTRIC COOPERATIVE, INC.
CITY OF TALLAHASSEE

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 21 to Facility Operating License No. DPR-72, issued to the Florida Power Corporation, City of Alachua, City of Bushnell, City of Gainesville, City of Kissimmee, City of Leesburg, City of New Smyrna Beach and Utilities Commission, City of New Smyrna Beach, City of Ocala, Orlando Utilities Commission and City of Orlando, Sebring Utilities Commission, Seminole Electric Cooperative, Inc., and the City of Tallahassee (the licensees) which revised the license and its appended Technical Specifications for operation for the Crystal River Unit No. 3 Nuclear Generating Plant (the facility) located in Citrus County, Florida. The amendment is effective as of the date of issuance.

This amendment revises the Technical Specifications to allow "racking out" the power supply breakers for the high pressure injection isolation valves during Mode 4 operation. These flow paths are isolated as part of the over-pressure mitigating system at the facility. This action satisfies the requirements of license condition 2.C.(6). This condition is therefore removed from the license.

The application for the amendment complies 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 § 51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated January 23, 1978, as supplemented July 3, 1979, (2) Amendment No. 21 to License No. DPR-72, 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 Crystal River Public Library, Crystal River, 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 3rd day of July 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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