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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. 20 to Facility Operating License No. DPR-72 for Crystal River Unit No. 3 Nuclear Generating Plant in response to your applications dated May 25 and July 3, 1979.

The amendment modifies the Technical Specifications by changing the required sodium hydroxide concentration in the reactor building spray chemical additive tank and by increasing the required shutdown margin during Modes 4 and 5. The action satisfies the requirements of license condition 2.C.(4). 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

Original signed by

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

Enclosures:
1. Amendment No. 20 to DPR-72
2. Safety Evaluation
3. Notice

cc w/encl:
See next page

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DATE	6/28/79	7/3/79	6/29/79	6/29/79	6/29/79	7/ /79

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Director, Power Production
Florida Power Corporation
P. O. Box 14042, Mail Stop C-4
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Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

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	RIngram	CNelson/cb	RReid	WGammill	S.H. Lewis	
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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. 20 to Facility Operating License No. DPR-72 for Crystal River Unit No. 3 Nuclear Generating Plant in response to your applications dated May 25 and July 3, 1979.

The amendment modifies the Technical Specifications by changing the required sodium hydroxide concentration in the reactor building spray chemical additive tank and by increasing the required shutdown margin during Modes 4 and 5. The action satisfies the requirements of license condition 2.C.(4). This condition is therefore removed from the license.

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Sincerely,

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

Enclosures:
1. Amendment No. 20 to DPR-72
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
P. O. Box 14042
St. Petersburg, Florida 33733

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

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Region IV Office
ATTN: EIS COORDINATOR
345 Courtland Street, N.E.
Atlanta, Georgia 30308

Director, Technical Assessment
Division
Office of Radiation Programs
(AW-459)
U. S. Environmental Protection Agency
Crystal Mall #2
Arlington, Virginia 20460

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Crystal River, Florida 32629

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dtd: 5/25/79
Bureau of Intergovernmental
Relations
660 Apalachee Parkway
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
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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. 20
License No. DPR-72

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

   A. The applications for amendment by Florida Power Corporation, et al (the licensees) dated May 25 and July 3, 1979, 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 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-72 is hereby amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and 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. 20 , are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

   B. Delete paragraph 2.C.(4) in its entirety.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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. 20

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

Replace the following pages of 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.

3/4 6-12

3/4 1-1

3/4 1-2

3/4 1-14

3/4 1-16

B 3/4 1-2

B 3/4 1-3

Add the following new pages:

3/4 1-2a

3/4 1-2b

3/4 1-16a

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months by verifying a total leak rate $\leq$ 6 gallons per hour for the system at:

1. Normal operating pressure or a hydrostatic test pressure of $\geq$ 190 psig for those parts of the system downstream of the pump suction isolation valve, and

2. $\geq$ 55 psig for the piping from the containment emergency sump isolation valve to the pump suction isolation valve.

d. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The spray additive system shall be OPERABLE with the spray additive tank containing at least a contained volume of between 11,190 and 12,010 gallons of solution containing between 105,000 and 120,000 ppm of sodium hydroxide (NaOH).

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the spray additive system inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the spray additive system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The spray additive system shall be demonstrated OPERABLE:

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

b. At least once per 6 months by:

1. Verifying the contained solution volume in the tank, and

2. Verifying the concentration of the NaOH solution by chemical analysis.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1.1.1 The SHUTDOWN MARGIN shall be $\geq$1% $\Delta k/k$.

APPLICABILITY: MODES 1, 2* and 3.

ACTION:

With the SHUTDOWN MARGIN <1% $\Delta k/k$, immediately initiate and continue boration at $\geq$10 gpm of 11,600 ppm boric acid solution or its equivalent, until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIRLMENTS

4.1.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be $\geq$1% $\Delta k/k$:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODES 1 or 2#, at least once per 12 hours, by verifying that regulating rod groups withdrawal is within the limits of Specification 3.1.3.5.

c. When in Mode 2## within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading by consideration of the factors of e. below, with the regulating rod groups at the maximum insertion limit of Specification 3.1.3.6.

#With K_{eff} $\geq$1.0.
##With K_{eff} <1.0.
*See Special Test Exception 3.10.4.

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODE 3, at least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. Control rod position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within +1% $\Delta k/k$ at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.1.e above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1.1.2 The SHUTDOWN MARGIN shall be $\geq$2.2% Δk/k.

APPLICABILITY: MODES 4 and 5.

ACTION:

With the SHUTDOWN MARGIN <2.2% Δk/k, immediately initiate and continue boration at $\geq$10 gpm of 11,600 ppm boric acid solution or its equivalent, until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.2.1 The SHUTDOWN MARGIN shall be determined to be $\geq$2.2% Δk/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

b. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading by consideration of the factors of c below, with the regulating rod groups at the maximum insertion limit of Specification 3.1.3.6.

SURVEILLANCE REQUIREMENTS (Continued)

c. When in MODE 4 or 5, at least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. Control rod position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within +1% $\Delta k/k$ at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.2.1.c above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.8 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A concentrated boric acid storage system and associated heat tracing with:

1. A minimum contained borated water volume of 6000 gallons,

2. Between 11,600 and 14,000 ppm of boron, and

3. A minimum solution temperature of 105°F.

b. The borated water storage tank (BWST) with:

1. A minimum contained borated water volume of 13,500 gallons,

2. A minimum boron concentration of 2270 ppm, and

3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving CORE ALTERATION or positive reactivity changes until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.8 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,

2. Verifying the contained borated water volume of the tank, and

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.9 Each of the following borated water sources shall be OPERABLE:

a. The concentrated boric acid storage system and associated heat tracing with:

1. A minimum contained borated water volume of 6000 gallons,

2. Between 11,600 and 14,000 ppm of boron, and

3. A minimum solution temperature of 105°F.

b. The borated water storage tank (BWST) with:

1. A contained borated water volume of between 415,200 and 449,000 gallons,

2. Between 2270 and 2450 ppm of boron, and

3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

MODES 1, 2 and 3:

a. With the concentrated boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1% $\Delta k/k$ at 200°F within the next 6 hours; restore the concentrated boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the borated water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODE 4:

a. With the concentrated boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 2.2% Δk/k at 200°F within the next 6 hours; restore the concentrated boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the borated water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 525°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, 3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and 4) the reactor pressure vessel is above its minimum RT_{NDT} 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) makeup or DHR pumps, 3) separate flow paths, 4) boric acid pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE emergency busses.

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 capacility requirement occurs at EOL from full power equilibrium xenon conditions and requires either 6000 gallons of 11,600 ppm boric acid solution from the boric acid storage tanks or 38,270 gallons of 2270 ppm borated water from the borated water storage tank.

The requirements for a minimum contained volume of 415,200 gallons of borated water in the borated water storge 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.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the

3/4.1.2 BORATION SYSTEMS (Continued)

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.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 2.2% of $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either 300 gallons of 11,600 ppm boric acid solution from the boric acid storage system or 1608 gallons of 2270 ppm borated water from the borated water storage tank. To envelop future cycle BWST contained borated water volume requirements, a minimum volume of 13,500 gallons is specified.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics. The limits on contained water volume, and boron concentration ensure a pH value of between 7.2 and 11.0 of the solution sprayed within containment after a design basis accident. The pH band minimizes the evolution of iodine and minimizes the effect of chlorids and caustic stress corrosion cracking on mechanical systems and components.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section (1) ensure that acceptable power distribution limits are maintained, (2) ensure that the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of a rod ejection accident. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met. For example, misalignment of a safety or regulating rod requires a restriction in THERMAL POWER. The reactivity worth of a misaligned rod is limited for the remainder of the fuel cycle to prevent exceeding the assumptions used in the safety analysis.

The position of a rod declared inoperable due to misalignment should not be included in computing the average group position for determining the OPERABILITY of rods with lesser misalignments.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON D C 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 20 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL

CRYSTAL RIVER UNIT NO. 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

Introduction

By application dated September 1, 1977, as replaced by application dated May 25, 1979, the Florida Power Corporation (FPC or the licensee) proposed changes to Specification 3.6.2.2, Spray Additive System, of the Technical Specifications of Crystal River Unit 3 Nuclear Generating Plant (CR-3). The licensee proposed a reduction in the acceptable concentration limits of sodium hydroxide in Tank BST-2 in the Chemical Additive System (CAS) of the CR-3 Reactor Building Spray System (RBSS). The licensee proposes to change the acceptable range of sodium hydroxide concentrations in Tank BST-2 from between 21.2 to 22.3 weight percent to between 10.5 to 12.0 weight percent. The range of acceptable values of the volume of sodium hydroxide in Tank BST-2 is not being changed.

Condition 2.C.(4) of the CR-3 operating license required the licensee to isolate the sodium thiosulfate tank and its contents from the RBSS CAS until permanent modifications to the CAS were submitted to NRC for review and approval. This is discussed in Supplement No. 3 dated December 1976 to the Safety Evaluation (SE) dated July 1974. The isolation of Tank BST-1 from the CAS and the deletion of specifications on this tank were reviewed and approved in the SE dated January 4, 1979. The proposed reduction in sodium hydroxide concentration limits for Tank BST-2 implement the licensee's permanent modifications to the CAS.

Evaluation

We have reviewed and evaluated the data provided by the licensee on the CR-3 RBSS in his letters dated September 1, 1977 and May 25, 1979. By letter dated May 25, 1979, the licensee showed that the pH of the CR-3 RBSS injection spray and recirculation (sump) spray water was between 7.9 and 11 and the potential consequences of the postulated loss of coolant accident (LOCA) were calculated to be less than the exposure guidelines of 10 CFR Part 100. The concentrations of sodium hydroxide in the CAS used in the licensee's calculations are the values proposed in the licensee's letter dated May 25, 1979.

On the basis of the data he has provided in his letter dated May 25, 1979, the licensee states that the CR-3 RBSS is adequate to assure acceptable spray water chemistry and potential consequences of the postulated LOCA which are less than the exposure guidelines of 10 CFR Part 100.

To show there is adequate assurance that sufficient sodium hydroxide will be added by gravity to the RBSS during a LOCA, the licensee has provided data from tests of the CAS. The licensee ran a series of measured drawdowns of the Borated Water Storage Tank, Tank BST-1 and Tank BST-2. These tests were made during the startup and test program of CR-3 in 1975 when Tank BST-1 was not isolated from the RBSS and contained sodium thiosulfate. After these water tests were performed, four spring-loaded stop check valves were replaced with conventional spring check valves and the CAS was retested. The tests were run pumping water from the tanks into the fuel transfer canal. These tests were compared to calculations made by the GAI computer program "Thermal Hydraulic Analysis." Comparison of the results are given in references (1) and (2). We conclude, based on these comparisons, that the computer program can accurately predict the performance of the CAS in the RBSS and can be used to determine the adequacy of the RBSS to maintain acceptable spray water pH.

Because the minimum pH of the RBSS spray water is less than 8.5, the evaluation of the CR-3 RBSS in Supplement No. 3 dated December 1976 of the SE dated July 1974 is no longer valid as it concerns (1) the RBSS spray water pH during the LOCA and (2) the potential consequences of the postulated LOCA. Our evaluation of the CR-3 RBSS during the LOCA concerning the above two items, based on the latest data on the CR-3 RBSS supplied by the licensee, is below. In addition, the evaluation in Supplement No. 3 above did not include the potential consequences due to leakage from safeguards equipment outside containment. This equipment outside containment, Decay Heat Removal and RBSS spray pumps and piping, circulates potentially highly radioactive water from the containment sump back to containment. Leakage from these systems outside containment will contribute to the potential consequences of the postulated LOCA. The potential consequences given in this evaluation will include a contribution from this pathway.

The requirements on post-accident spray water chemistry are discussed in Standard Review Plan (SRP) 6.5.2. The pH of the RBSS spray water during injection and recirculation should be between 8.5 and 11.0. As given in the Tables in the licensee's May25, 1979 letter, the pH of the RBSS spray during the LOCA is between 7.9 and 11.0. This includes the case of failure of one of the two valves at Tank BST-2 and all of the sodium hydroxide from the tank entering one of two operating RBSS trains which results in the maximum pH in the spray. For the most restrictive single active failure in the RBSS, in terms of the potential consequences of the postulated LOCA, the loss of a spray pump, the minimum pH of the RBSS spray is 8.1.

The minimum pH of the RBSS spray water is below 8.5 because the maximum pH was not allowed to be greater than 11.0. The requirement that the maximum spray water pH must be less than 11.0 is more important than the requirement that the minimum pH should be 8.5 or greater. Keeping the spray water pH no greater than 11 will prevent caustic stress corrosion cracking and degradation of the RBSS piping during a LOCA. Keeping the spray water pH no less than 8.5 will inhibit chloride stress corrosion cracking and degradation of the RBSS piping and will provide maximum spray effectiveness at preventing radioiodines released to the containment atmosphere during a LOCA from being released outside to the environment. Allowing the spray water pH to be less than 8.5 will reduce the effectiveness of the spray water to remove radioiodine from the containment atmosphere and to retain this radioiodine in the water. However, with the pH of the RBSS spray water above 7, the water will still inhibit chloride stress corrosion cracking in the RBSS piping and will still remove radioiodine from the containment atmosphere and retain it in the water. The effectivness of the spray to remove radioiodines from the containment atmosphere and retain it in the water and, thus, to reduce the potential consequences of the postulated LOCA increases rapidly with pH between 7 and 8.5.

We have calculated the potential consequences of the postulated LOCA at CR-3 with the proposed changes to the RBSS CAS. This is for a minimum RBSS spray water pH of 8. The potential consequences and the assumptions made to calculate them are given in Table 1 and Table 2, respectively. The potential consequences of the postulated LOCA are well within the exposure guidelines of 10 CFR Part 100. The potential consequences include a contribution due to leakage from safeguard equipment located outside containment. This contribution to potential consequences had not been included in previous evaluations of the postulated LOCA; however, the licensee does have specifications limiting the maximum acceptable leakage from this equipment outside containment. Compliance with Specification 4.5.2.e.5 (Decay Heat Removal System) and 4.6.2.1.b (RBSS) provide assurance that the leakage rates assumed for the two systems during the postulated LOCA will not be exceeded. Because the potential consequences of the postulated LOCA are within the exposure guidelines of 10 CFR Part 100, the potential consequences are acceptable and the proposed Specification 3.6.2.2 is acceptable as written.

Moderator Dilution

Based on an operating experience it had been determined that inadvertent injection of the contents of BST-2 into the reactor coolant system must be considered as a potential moderator dilution accident. Currently procedural restrictions are imposed to preclude this accident while the issue is being reviewed. However, during review of the spray additive system discussed above, it was determined that the potential for boron dilution by this means existed at times during Mode 4 operation with the decay heat system lined up for recirculation.

By letter dated July 3, 1979, the licensee proposed a Technical Specification change which would require a shutdown margin of $\geq$2.2% $\Delta k/k$ instead of $\geq$1.0% $\Delta k/k$ during Mode 4. This requirement was also proposed for mode 5 for consistency of operation. In support of this change the licensee presented an analysis which demonstrates that with this shutdown margin the operator has at least 15 minutes to take action before the reactor core could become critical. This meets the current criteria in Standard Review Plan 15.4.6.

We have reviewed the licensee's analysis and proposed changes to the Technical Specifications and find them acceptable.

Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types, an increase in total amounts of effluents or an increase in power level and therefore 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 conclude on the basis of the above considerations that the proposed changes to Specification 3.6.2.2 of the CR-3 Technical Specifications is acceptable as written.

We also have concluded, based on the considerations 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.

References:

1. Ely, R. F. Jr., "Borated Water Storage Tank Drawdown Transient Analysis, Revision 1," for Crystal River Unit 3 Nuclear Generating Plant, Florida Power Corporation, 30 January 1976.

2. Ely, R. F. Jr., "Hydraulic Analysis of Piping Networks Using PIPF Computer Program," Topical Report GAI-TR-105, December, 1976.

Dated: July 3, 1979

TABLE 1

POTENTIAL OFFSITE DOSES OF THE POSTULATED LOSS-OF-COOLANT ACCIDENT

Accident	Two Hour Exclusion Boundary (1340 Meters)		Course of Accidents Low Population Zone (8047 Meters)	
	Thyroid (Rem)	Whole Body (Rem)	Thyroid (Rem)	Whole Body (Rem)
Loss of Coolant				
Leakage thru containment	150.	3.0	27.	0.7
Leakage outside containment	4.8	.012	4.1	.004
Post-LOCA Hydrogen Purge Dose	-	-	<1	<1

TABLE 2

ASSUMPTIONS FOR THE POSTULATED LOSS-OF-COOLANT ACCIDENT

Hydrogen Purge Dose Analysis

Using Regulatory Guide 1.7 assumptions, the licensee has calculated a hydrogen purge dose of approximately 0.1 Rem at the Low Population Zone. Our independent calculations are in substantial agreement with this incremental dose.

Loss-of-Coolant Accident

Regulatory Guide 1.4, Revision 2, 1974

Power (MWt)	2544
Containment Volume (ft^3)	2×10^6
Volume In Sump (lbs.)	3.8×10^6
Distribution of Radioiodines (%)	
elemental	91
organic	4
particulate	5

Through Containment Leakage

Design Containment Leak Rate (%/day)	
0 - 24 hours	0.25
24 hours	0.125
Spray fall height (ft)	96
Spray flow rate (gpm)	1500
Spray reduction limits	
elemental iodine	50
Partition Coefficient	
elemental iodine	1600 (pH=8)
Spray Removal rates (hrs^{-1})	
elemental iodine	7.05
organic iodine	0.0
particulate iodine	0.45
Unsprayed region (%)	25.

TABLE 2 (Cont'd)

Leakage Outside Containment (gph)	
Reactor Building Spray System	12
Decay Heat Removal System	12
Charcoal Filter Efficiency (%)	
elemental iodine	90
organic iodine	70
particulate iodine	90
Percent of Iodine Released (%)	10
Start of Recirculation After LOCA (Hr)	0.67
Atmospheric Dispersion Factors (sec/m^3)	
0 - 2 hours (1340 meters)	2.2×10^{-4}
0 - 8 hours (8047 meters)	1.0×10^{-5}
8 - 24 hours	6.8×10^{-6}
24 - 96 hours	2.8×10^{-6}
96 - 720 hours	7.5×10^{-7}

UNITE STATES NUCLEAR REGULATORY COMMI ON

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. 20 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 modifies the Technical Specifications by changing the required sodium hydroxide concentration in the reactor building spray chemical additive tank and by increasing the required shutdown margin during Modes 4 and 5. The action satisfies the requirements of license condition 2.C.(4). 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

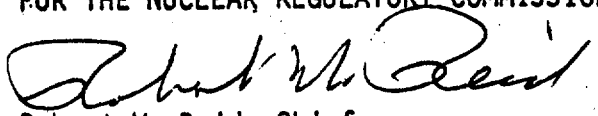
Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve 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 applications for amendment dated May 25 and July 3, 1979, (2) Amendment No. 20 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, Chief
Operating Reactors Branch #4
Division of Operating Reactors