DMB 016

May 25, 1984

Docket No. 50-302

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Mr. Walter S. Wilgus
Vice President, Nuclear Operations
Florida Power Corporation
ATTN: Manager, Nuclear Licensing
& Fuel Management
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St. Petersburg, Florida 33733

Dear Mr. Wilgus:

The Commission has issued the enclosed Amendment No. 68 to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant (CR-3). This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated January 30, 1984, and as supplemented on March 20, 1984.

This amendment raises the Crystal River Unit 3 Technical Specification limit on maximum steam generator secondary water level from 360 inches to approximately 380 inches in operating modes 1 through 3 to provide more operating flexibility. This change was justified by supplementary analyses performed on behalf of Florida Power Corporation and submitted to the Commission.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's next Monthly Federal Register Notice.

Sincerely,

"ORIGINAL SIGNED BY:"
George W. Rivenbark, Acting Chief
Operating Reactors Branch No. 4
Division of Licensing

Enclosures:
1. Amendment No. 68 to DPR-72
2. Safety Evaluation

cc w/enclosures:
See next page

ORB#4:DL	ORB#4:DL	C-ORB#4:DL	AD/OR:DL	OELD	RSB/DSI
RIngram	RHernan;ef	GRivenbark	GLainas	M. Karman	BSheron
05/23/84	05/22/84	05/22/84	05/25/84	05/23/84	05/25/84

SER ok
As marked for issue at 30 days (NSHC not needed)

remove indicator under evaluation section.

Crystal River Unit No. 3
Florida Power Corporation

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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
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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. 68
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 30, 1984 (as supplemented on March 20, 1984), 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, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-72 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 68, are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

George W. Rivenbark

George W. Rivenbark, Chief
Operating Reactors Branch No. 4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 25, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 68

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

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

3/4 4-6

3/4 4-10a (new page)

B 3/4 4-3

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with:

a. A steam bubble,

b. A water level between 40 and 290 inches, and

c. At least 126 kW of pressurizer heaters.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the pressurizer inoperable, be in at least HOT STANDBY with the control rod drive trip breakers open within 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 The pressurizer shall be demonstrated OPERABLE by verifying pressurizer level to be within limits at least once per 12 hours.

4.4.4.2 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by manually transferring power from the normal to the emergency power supply and energizing the heaters.

REACTOR COOLANT SYSTEM

STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE with a maximum water level as specified on Figure 3.4-5 and a minimum water level of 18 inches.

APPLICABILITY: MODES 1, 2, 3 and 4.*

ACTION:

a. With one or more steam generators inoperable due to steam generator tube imperfections, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

b. With one or more steam generators inoperable due to the water level being outside the limits, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

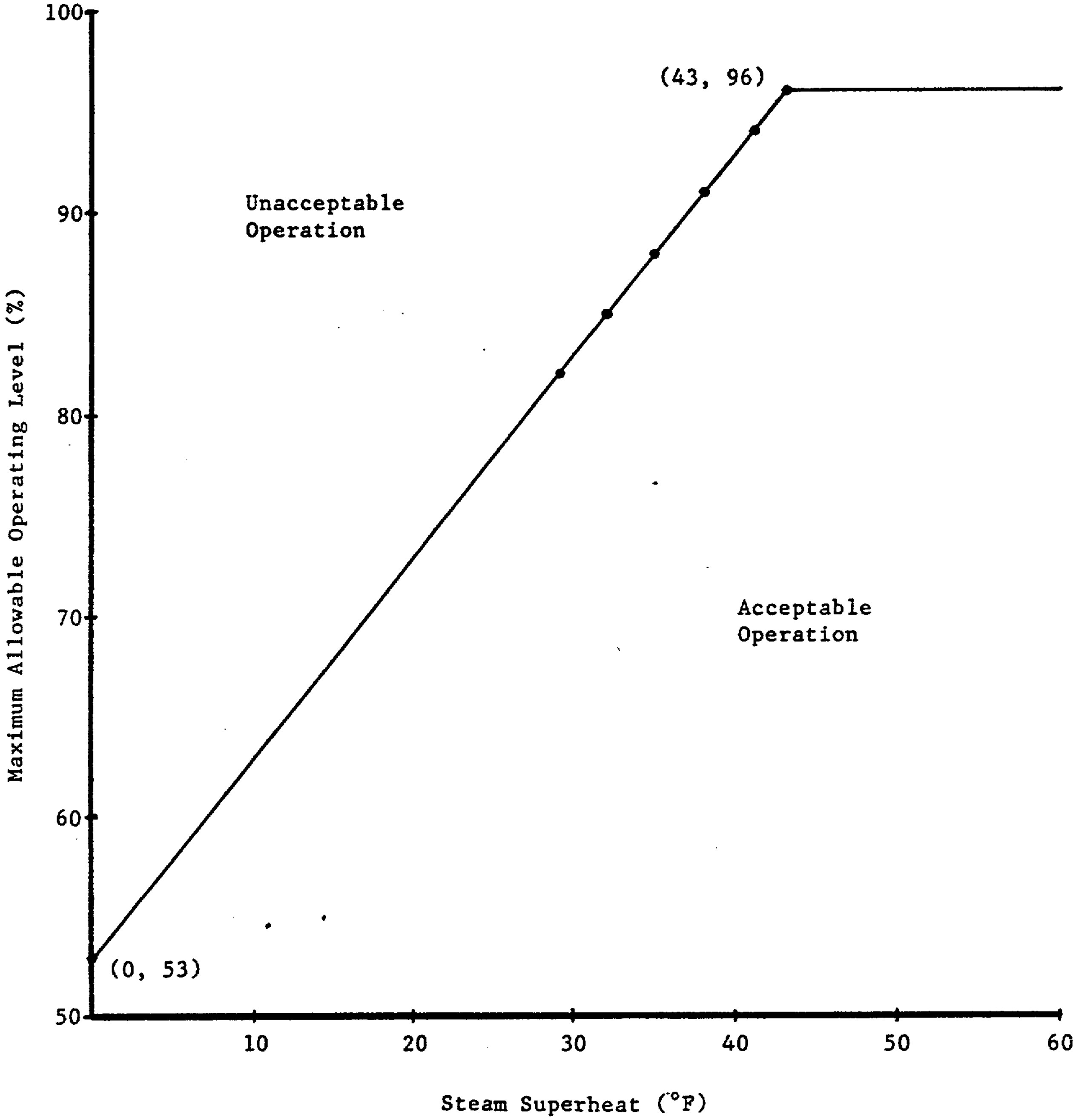
4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

*As noted on Figure 3.4-5, maximum water level restrictions are not applicable in MODE 4.

FIGURE 3.4-5

MAXIMUM ALLOWABLE STEAM GENERATOR LEVEL*



*Not applicable to Mode 4 operation.

system and the secondary coolant system (primary-to-secondary leakage = 1 GPM). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 1 GPM can be detected by monitoring the secondary coolant. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Operational experience has shown that tube defects can be the result of unique operating conditions and/or physical arrangements in specific limited areas of the steam generators (for example, tubes adjacent to the open inspection lane or tubes whose 15th tube support plate hole is not broached but drilled). A full inspection of all of the tubes in such specific limited areas will provide complete assurance that degraded or defective tubes in these areas are detected. Because no credit is taken for these distinctive tubes in the constitution of the first sample or its results, the requirements for the first sample are unchanged. This requirement is essentially equivalent to and meets the intent of the requirements set forth in Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes", Rev. 1, July 1975, and does not reduce the margin of safety provided by those requirements.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

The steam generator water level limits are consistent with the initial condition assumptions in the FSAR. The maximum steam generator level, as a function of steam superheat, is sufficient to assure a mass inventory of less than or equal to 62,600 lbm in the event of a main stream line rupture.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. (These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.)

3/4.4.6.2 OPERATIONAL LEAKAGE

Industry experience has shown that, while a limited amount of leakage is expected from the RCS, the UNIDENTIFIED LEAKAGE portion of this can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The CONTROLLED LEAKAGE limit of 10 GPM restricts operation with a total RCS leakage to all RC pump seals in excess of 10 GPM.

The total steam generator tube leakage limit of 1 GPM for all steam generators ensures that the dosage contribution from tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 GPM limit is consistent with the assumptions used in the analysis of these accidents.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 68 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 letter dated January 30, 1984, and as supplemented on March 20, 1984, Florida Power Corporation (FPC or the licensee) proposed a change to the Crystal River Unit 3 (CR-3) Technical Specifications (TSs). This change request was to raise the maximum operating steam generator level limit to allow greater operating flexibility and to allow achievement of rated power level when the present level limit becomes restricting due to mechanical conditions such as tube support plate fouling.

Background

The downcomer water level within the Once Through Steam Generators (OTSG) at Crystal River Unit 3 has been increasing at a rate of about 1% per month during full power operation.

The cause of the level increase is attributed to fouling in the broached hole area of the tube support plates. The fouling causes an increased resistance to steam flow within the tube region which causes the water level to rise within the downcomer region.

In order to remain within the TS requirement for downcomer water level, operation of Crystal River Unit 3 is currently limited to 96% power. The licensee requests that the TS limit be increased to permit full power operation.

The current Technical Specification limit for downcomer water level is 360 inches which corresponds to about 87% on the operating range scale. The licensee requests that the limit be raised to 96% on the operating range which corresponds to approximately 382 inches of downcomer level above the lower tube sheet. The aspirator ports within the steam generator shroud which provide for feedwater heating are located at 386 inches. The original limit was based on a conservative calculation of level at the maximum steam generator inventory assumed in the Final Safety Analysis Report (FSAR) of 62,600 lbm. This is the steam generator inventory assumed for the main steam line break calculation of reactor overcooling, environmental doses and containment pressure. The licensee submitted a new calculation of steam generator inventory which concludes the inventory will be less than 57,000 lbm even at the new level limit. This result was confirmed by the staff using the RELAP5 computer code.

Although no loss of superheat has occurred at Crystal River indicating no tube surface fouling, the licensee's proposed change includes a level penalty which would be applied in the event that tube fouling occurs in future operations.

Evaluation

The staff concludes that operation at power with up to 96% level on the operating range is acceptable and that the licensee's proposed TS is acceptable as discussed in more detail below.

The licensee also requests that the TSs be revised to remove steam generator level requirements for Mode 4 operation (hot shutdown). Currently the steam generators are allowed to be filled only at cold shutdown. The change would allow filling of the OTSGs at temperatures up to 280°F. The consequences of a main steam line rupture at this temperature would be less than that of the FSAR case at full power. The staff concludes the licensee's proposed change removing level restrictions for Mode 4 operation is acceptable.

The most limiting design-basis accident which would be affected by steam generator operating level, and hence the limit addressed by this amendment, is a steam line failure. This accident is evaluated in Section 14.2.2.1 of the Crystal River FSAR and in the staff's Safety Evaluation Report. The parameter of interest is the mass of water, or inventory, contained in the OTSG due to its role in lowering reactor coolant system temperature and in raising containment pressure during a steam line break accident. A higher inventory causes the effects of the accident to be more severe. The FSAR assumes an inventory of 62,600 lbm for the purpose of analyzing this accident, which was conservatively high and doesn't account for the dynamic complexity of the fluid (i.e. mixture of water and steam) in the OTSG. In order to reevaluate the present OTSG maximum water level limit, the licensee had a more accurate and realistic analysis performed by Babcock & Wilcox which demonstrated that this limit should logically be established by a curve of level vs. superheat rather than a fixed numerical value of level. If the water level should tend to rise above the 96% upper limit, the superheat would rapidly tend to decrease sharply, requiring a reduction in water level. Thus, the superheat vs. level limitation also tends to assure that, in normal operation, water level will remain clear of the aspirator ports. Hence, the licensee has proposed such a curve which will establish the regime of allowed values of level for a given amount of superheat, which is a more accurate prediction of actual water inventory in the OTSG. The curve proposed by the licensee is based upon maintaining inventory less than 57,000 lbm which is 10% less than the inventory used in the FSAR accident analysis and is, therefore, more conservative than the original analysis. Therefore, the proposed limit falls well within the original analysis and would therefore not involve an increase in the consequences of an accident previously evaluated. Since the analysis for the steam line break accident does not assume failure of the OTSG itself, this change does not have any affect on the probability of this accident. The licensee also requested to delete applicability of an upper OTSG limit to Mode 4 of operation. Because the severity of a steam line break accident drops sharply as OTSG temperature and pressure drop, and because temperature is limited to a maximum of 280°F in Mode 4, the staff considers that this change is acceptable.

The TS changes proposed by the licensee and incorporated by this amendment involve only an incremental increase (6%) in maximum OTSG level, and in fact could impose lower level limits with a lower value of steam superheat. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. Rather, it imposes a more accurate and realistic method of limiting maximum OTSG inventory for routine operation and accidents that have been previously evaluated in a great amount of detail.

In the most extreme set of allowed conditions proposed by the licensee (maximum allowable power level, OTSG level and temperature , minimum steam superheat), the OTSG inventory shown by analysis is approximately 10% below that assumed by the FSAR analysis for the steam line break analysis. With the new limit, the allowable OTSG level (and hence inventory) would actually be less than presently allowed for reduced values of steam superheat. Therefore, although the margin of safety may be incrementally reduced in some cases and incrementally increased in others compared to the present limit, in all cases, the margin is at least 10% below the FSAR analysis.

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) 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.

Dated: May 25, 1984

Principal Contributors:
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