July 17, 1986

DMB 014

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
P. O. Box 14042; M.A.C. H-3
St. Petersburg, Florida 33733

Dear Mr. Wilgus:

The Commission has issued the enclosed Amendment No. 90 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 April 23, 1984.

This amendment revises immediate reporting requirements and incorporates the new reporting system for significant events at nuclear power plants. These changes result from changes to NRC regulations, 10 CFR 50.72 and 50.73, which became effective January 1, 1984.

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

Sincerely,

Brenda Mozafari, Project Manager
PWR Project Directorate #6
Division of PWR Licensing-B

Enclosures:
1. Amendment No. 90 to DPR-72
2. Safety Evaluation

cc w/enclosures:
See next page

Raw for	BRM		Raw	AZ Edison 7/3	OGC MKarman/ri
PBD-6	PBD-6	PBD-6	PBD-6	~~PBD-6~~	~~OELD~~
RIngram	BMozafari;cf	HSilver	RWeller	JStolz	
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Mr. W. S. Wilgus
Florida Power Corporation

Crystal River Unit No. 3 Nuclear Generating Plant

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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
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. 90
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 April 23, 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. 90 , 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

John F. Stolz

John F. Stolz, Director
PWR Project Directorate #6
Division of PWR Licensing-B

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: July 17, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 90

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

Replace the following pages of the Appendix "A" Technical Specifications with the attached 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.

Remove	Insert
I	I
XVI	XVI
1-2	1-2
3/4 3-40	3/4 3-40
3/4 4-10	3/4 4-10
3/4 4-12	3/4 4-12
3/4 4-20	3/4 4-20
3/4 6-9a	3/4 6-9a
3/4 7-38	3/4 7-38
3/4 7-41	3/4 7-41
3/4 7-44	3/4 7-44
6-6	6-6
6-10	6-10
6-11	6-11
6-12	6-12
6-13	6-13
6-15	6-15
6-16	6-16
6-17	6-17
6-17a	--
6-18	6-18

DEFINITIONS

SECTION | PAGE

ADMINISTRATIVE CONTROLS

1.0 DEFINITIONS

DEFINED TERMS

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

THERMAL POWER

1.2 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2544 MWt.

OPERATIONAL MODE

1.4 An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principle specification and shall be part of the specifications.

OPERABLE - OPERABILITY

1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment, that are required for the system, subsystem, train, component or device to perform its function(s), are also capable of performing their related support function(s).

REPORTABLE EVENT:

1.7 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

CONTAINMENT INTEGRITY

1.8 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE containment automatic isolation system, or

2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.1.

b. All equipment hatches are closed and sealed,

c. Each airlock is OPERABLE pursuant to Specification 3.6.1.3,

d. The containment leakage rates are within the limits of Specification 3.6.1.2, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

CHANNEL CALIBRATION

1.9 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.10 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: Whenever equipment in that fire detection zone is required to be OPERABLE.

ACTION:

With one or more of the fire detection instrument(s) shown in Table 3.3-11, inoperable:

a. Within 1 hour, establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, and

b. Restore the inoperable instrument(s) to OPERABLE status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the malfunction and the plans and schedule for restoring the instrument(s) to OPERABLE status.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 Each of the above fire detection instruments shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST.

4.3.3.7.2 The circuitry associated with the detector alarms listed in Table 3.3-11 shall be demonstrated OPERABLE at least once per 6 months for all National Fire Protection Association (NFPA) Code 72D Class B supervised circuits.

4.3.3.7.3 The non-supervised circuits between the local panels and the control room for the detectors listed in Table 3.3-11 shall be demonstrated OPERABLE at least once per 31 days.

or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections ≥20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective. Any tube which does not permit the passage of the eddy-current inspection probe shall be deemed a defective tube.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service because it may become unserviceable prior to the next inspection and is equal to 40% of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3.c, above.

8. Tube Inspection means an inspection of the entire steam generator tube as far as possible.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2 (and Table 4.4-6, if the provisions of Specification 4.4.5.2.d are utilized).

4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

SURVEILLANCE REQUIREMENTS (Continued)

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported to the Commission pursuant to 10 CFR 50.72 prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence pursuant to 10 CFR 50.73.

4.4.5.6 The steam generator shall be demonstrated OPERABLE by verifying steam generator level to be within limits at least once per 12 hours.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	Yes
No. of Steam Generators per Unit	Two
First Inservice Inspection	One
Second & Subsequent Inservice Inspections	One[1]

Table Notation:

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 6% of the tubes if the results of the first or previous inspections indicate that both steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one steam generator may be found to be more severe than those in the other steam generator. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

TABLE 4.4-2
STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes.
			C-3	Perform action for C-3 result of first sample	C-3	Perform action for C-3 result of first sample.
					N/A	N/A
	C-3	Inspect all tubes in this S.G., plug defective tubes, and inspect 2S tubes in each other S.G. Notify NRC pursuant to 10 CFR 50.72.	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no add-itional S.G.s are C-3	Perform action for C-2 result of second sample.	N/A	N/A
			Addi-tional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes. Notify NRC pursuant to 10 CFR 50.72.	N/A	N/A

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit and n is number of steam generators inspected during an inspection.

TABLE 4.4-3

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETER	SAMPLE AND ANALYSIS FREQUENCY
DISSOLVED OXYGEN*	At least once each 72 hours
CHLORIDE	At least once each 72 hours
FLUORIDE	At least once each 72 hours

*Not required with $T_{avg} \leq 250°F$.

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the primary coolant shall be limited to:

a. $\leq$ 1.0 μCi/gram DOSE EQUIVALENT I-131, and

b. $\leq$ 100/$\bar{E}$ μCi/gram

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1, 2 and 3*

a. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.4-1, operation may continue for up to 48 hours provided that operation under these circumstances shall not exceed 10% of the unit's total yearly operating time. The provisions of Specification 3.0.4 are not applicable.

b. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} < 500°F within 6 hours.

c. With the specific activity of the primary coolant > 100/$\bar{E}$ μCi/gram, be in at least HOT STANDBY with T_{avg} < 500°F within 6 hours.

MODES 1, 2, 3, 4 and 5:

a. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 or > 100/$\bar{E}$ μCi/gram, perform the sampling and analysis requirements of item 4 a) of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within the next 30 days. This report shall contain the results of the specific activity analyses together with the following information:

*With $T_{avg} \geq$ 500°F

SURVEILLANCE REQUIREMENTS (Continued)

b. Removing one wire or strand from each of a dome, vertical and hoop tendon checked for lift off force and determining that over the entire length of the removed wire or strand that:

1. The tendon wires or strands are free of corrosion, cracks and damage,

2. There are no changes in the presence or physical appearance of the sheathing filler grease, and

3. A minimum tensile strength value of 240,000 psi (guaranteed ultimate strength of the tendon material) for at least three wire or strand samples (one from each end and one at mid-length) cut from each removed wire or strand. Failure of any one of the wire or strand samples to meet the minimum tensile strength tests is evidence of abnormal degradation of the containment structure.

4.6.1.6.2 End Anchorages and Adjacent Concrete Surfaces The structural integrity of the end anchorages of all tendons inspected pursuant to Specification 4.6.1.6.1 and the adjacent concrete surfaces shall be demonstrated by determining through inspection that no apparent changes have occurred in the visual appearance of the end anchorage or the concrete crack patterns adjacent to the end anchorages.

4.6.1.6.3 Containment Surfaces The structural integrity of the exposed accessible interior and exterior surfaces of the containment, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test (Specification 4.6.1.2) by a visual inspection of these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.5.1.6.4 Containment Dome The containment dome's structural integrity shall be demonstrated at the end of 1 year, 18 months, 2 years, 3 years, 40 ±10 months (coincident with the first periodic integrated containment leak rate test), and 5 years following the initial containment structural integrity test. The dome's structural integrity shall be demonstrated by:

a. Measuring the elevation difference of 7 dome survey points (1 at the apex; 3 at a radius of $\simeq$ 29 feet at azimuths 90°, 215° and 334°; and 3 at a radius of $\simeq$ 49 feet at azimuths 90°, 215° and 334°) and 3 benchmarks (on Ring Girder at azimuths 90°, 215° and 334°) along the respective azimuths. These elevation differences shall be compared to the elevation differences established by the Baseline Survey. If the containment is in a normal operation/shutdown mode, the acceptable change in elevation differences will be based on consideration of expected movement and survey accuracy coupled with an acceptable strain level for the radial reinforcement. Changes of a greater magnitude shall require an engineering evaluation. If the containment is in a pressurized mode for a periodic containment integrated leak rate test, the acceptable changes in elevation differences will be similar to that for the initial containment structural integrity test applied to the elevation differences during the periodic containment integrated leak rate test.

b. Measuring crack widths and plotting crack patterns in the area of the dome 3 feet on either side of azimuths 195° from the apex to the Ring Girder. Cracks wider than 0.010 inches will be plotted and cracks wider than 0.040 inches shall require an engineering evaluation. In addition, a general visual inspection of the entire dome surface area shall be performed.

4.6.1.6.5 Reports Any abnormal degradation of the containment structure detected during the above required tests and inspections shall be reported to the Commission pursuant to 10 CFR 50.73. This report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective actions taken.

SURVEILLANCE REQUIREMENTS (Continued)

1. With a half-life greater than 30 days (excluding Hydrogen 3) and

2. In any form other than gas.

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.10.1.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of $\geq$ 0.005 microcuries of removable contamination.

3/4 7.11 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.11.1 The fire suppression water system shall be OPERABLE with:

a. At least two high-pressure pumps each with a capacity of 2000 gpm, with their discharge aligned to the fire suppression header.

b. Separate water supplies, each with a minimum contained water volume of 345,000 gallons.

c. An OPERABLE flow path capable of taking suction from the water supply and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves and the front valve ahead of the water flow alarm device on each sprinkler, hose standpipe and spray system riser required to be OPERABLE per Specifications 3.7.11.2 and 3.7.11.4.

APPLICABILITY: At all times.

ACTION:

a. With one pump and/or one water supply inoperable, restore the inoperable equipment to OPERABLE status within 7 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the plans and procedures to be used to provide for the loss of redundancy in this system. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

b. With the fire suppression water system otherwise inoperable:

1. Establish a backup fire suppression water system within 24 hours, and

2. Submit a Special Report in accordance with Specification 6.9.2:

    a. By telephone within 24 hours,

    b. Confirm by telegraph, mailgram, or facsimile transmission no later than the first working day following the event, and

    c. In writing within 14 days following the event, outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

DELUGE AND SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.11.2 The deluge and sprinkler systems shown in Table 3.7-4 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the deluge/sprinkler protected areas is required to be OPERABLE.

ACTION:

a. With one or more of the above required deluge and sprinkler systems inoperable, establish a continuous fire watch with backup fire suppression equipment for the unprotected area(s) within 1 hour; restore the system to OPERABLE status within 14 days, or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

b. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.2 Each of the above required deluge and sprinkler systems shall be demonstrated OPERABLE:

a. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

b. At least once per 18 months:

1. By performing a system functional test which includes simulated automatic actuation of the system, and;

2. Verifying that the automatic valves in the flow path actuate to their correct positions.

b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

2. By inspection of the deluge headers to verify their integrity, and

3. By inspection of each nozzle to verify no blockage.

c. At least once per 3 years by performing an air flow test through each open head deluge/sprinkler header and verifying each open head deluge/sprinkler nozzle is unobstructed.

TABLE 3.7-4

DELUGE AND SPRINKLER SYSTEMS

SYSTEM	LOCATION
1. AHFL-1 Deluge System	Auxiliary Building, Elevation 143'0", Zone 11 (Main Filter Exhaust Room)
2. AHFL-2A Deluge System	Auxiliary Building, Elevation 143'0", Zone 11 (Main Filter Exhaust Room)
3. AHFL-2B Deluge System	Auxiliary Building, Elevation 143'0", Zone 11 (Main Filter Exhaust Room)
4. AHFL-2C Deluge System	Auxiliary Building, Elevation 143'0", Zone 11 (Main Filter Exhaust Room)
5. AHFL-2D Deluge System	Auxiliary Building, Elevation 164'0", Zone 5 (Main Filter Exhaust Room)
6. AHFL-4A Deluge System	Control Complex, Elevation 164'0", Zone 5 (HVAC Emergency Equipment 3A)
7. AHFL-4B Deluge System	Control Complex, Elevation 164'0", Zone 4 (HVAC Emergency Equipment 3B)
8. AHFL-15 Deluge System	Auxiliary Building, Elevation 95'0", Zone 34 (Nuclear Sampling Room)
9. Emergency Diesels Deluge	Auxiliary Building, Elevation 119'0", Zones 27 and 28

HALON SYSTEM

LIMITING CONDITON FOR OPERATION

3.7.11.3 The Halon system in the Cable Spreading room (Control Complex, Elevation 134'0") shall be OPERABLE with the storage tanks having at least 95% of full charge weight and 90% of full charge pressure.

APPLICABILITY: At all times.

ACTION:

a. With the above required Halon system inoperable, establish a continuous fire watch with backup fire suppression equipment for the unprotected area within 1 hour; restore the system to OPERABLE status within 14 days or, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.3 The Halon system shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying each Halon storage tank weight and pressure.

b. At least once per 18 months by

1. Verifying the system, including associated ventilation dampers, would actuate automatically to a simulated test signal.

2. Verifying the OPERABILITY of the manual initiating system.

3. Performance of a flow test through headers and nozzles to assure no blockage.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Chemistry and Radiation Protection Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and the Operations Technical Advisor, who shall have a Bachelor's degree, or the equivalent, in a scientific or engineering discipline with specific training in plant design and response and analysis of the plant for transients and accidents.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Nuclear Plant Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Nuclear Plant Training Manager and shall meet or exceed the requirements of Section 27 of the NFPA Code-1976, except for Fire Brigade training sessions which shall be held at least quarterly.

6.5 REVIEW AND AUDIT

6.5.1 PLANT REVIEW COMMITTEE (PRC)

FUNCTION

6.5.1.1 The Plant Review Committee shall function to advise the Nuclear Plant Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The Plant Review Committee shall be composed of the:

Chairman:	Technical Services Superintendent
Member:	Operations Superintendent
Member:	Maintenance Superintendent
Member:	Nuclear Technical Services Superintendent (Security)
Member:	QA/QC Compliance Manager
Member:	Chem/Rad Protection Manager
Member:	Technical Support Engineer
Member:	Performance Engineering Supervisor
Member:	At Large (Designated by Chairman)
Member:	At Large (Designated by Chairman)

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PRC Chairman to serve on a temporary basis; no more than two alternates shall participate as voting members in PRC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PRC shall meet at least once per calendar month and as convened by the PRC Chairman or his designated alternate.

QUORUM

6.5.1.5 A quorum of the PRC shall consist of the Chairman or his designated alternate and five members including alternates.

RESPONSIBILITIES

6.5.1.6 The Plant Review Committee shall be responsible for:

a. Review of 1) all procedures and changes thereto as required by Specification 6.8.2, 2) any other proposed procedures or changes thereto as determined by the Nuclear Plant Manager to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to the Appendix "A" Technical Specifications.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety, and changes to radwaste systems which could significantly alter their ability to meet Appendix I.

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Vice President, Nuclear Operations and to the Chairman of the Nuclear General Review Committee.

f. Review of all REPORTABLE EVENTS.

g. Review of facility operations to detect potential nuclear safety hazards.

h. Performance of special reviews, investigations, or analyses and reports thereon as requested by the Chairman of the Nuclear General Review Committee.

i. Review of the Plant Security Plan and implementing procedures.

j. Review of the Emergency Plan and implementing procedures.

CONSULTANTS

6.5.2.5 Consultants shall be utilized as determined by the NGRC Chairman to provide expert advice to the NGRC.

MEETING FREQUENCY

6.5.2.6 The NGRC shall meet at least once per calendar quarter during the initial year of facility operation following fuel loading and at least once per six months thereafter.

QUORUM

6.5.2.7 A quorum of NGRC shall consist of the Chairman or his designated alternate and five additional NGRC members, including alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

REVIEW

6.5.2.8 The NGRC shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes in Technical Specifications or this Operating License.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

Correction letter of 8-7-84

REVIEW (Continued)

g. All REPORTABLE EVENTS.

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

i. Reports and meetings minutes of the Plant Review Committee.

j. Changes to the PROCESS CONTROL PROGRAM and to the OFFSITE DOSE CALCULATION MANUAL.

AUDITS

6.5.2.9 Audits of facility activities shall be performed under the cognizance of the NGRC. These Audits shall encompass:

a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training and qualifications of the entire facility staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.

e. The Facility Emergency Plan and implementing procedures at least once per 24 months.

f. The Facility Security Plan and implementing procedures at least once per 24 months.

g. The Facility fire protection program and implementing procedures at least once per 24 months.

h. The radiological environmental monitoring program and the results thereof at least once per 12 months.

AUDITS (Continued)

i. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months.

j. The PROCESS CONTROL PROGRAM and implementing procedures for solidification of radioactive wastes at least once per 24 months.

k. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring at least once per 12 months.

l. Any other area of facility operation considered appropriate by the NGRC or the Senior Vice President-Engineering and Construction.

AUTHORITY

6.5.2.10 THe NGRC shall report to and advise the Senior Vice President-Engineering and Construction on those areas of responsibility specified in Sections 6.5.2.8 and 6.5.2.9.

RECORDS

6.5.2.11 Records of NGRC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each NGRC meeting shall be prepared, approved and forwarded to the Senior Vice President - Engineering and Construction, within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.8 above, shall be prepared, approved and forwarded to the Senior Vice President - Engineering and Construction, within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.9 above, shall be forwarded to the Senior Vice President - Engineering and Construction, and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR 50, and

b. Each REPORTABLE EVENT shall be reviewed by the PRC and submitted to the NGRC and the Vice President, Nuclear Operations.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The facility shall be placed in at least HOT STANDBY within one hour.

b. The Safety Limit violation shall be reported to the Commission, the Vice President, Nuclear Operations, and to the NGRC within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PRC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the NGRC and the Vice President,Nuclear Operations within 14 days of the violation. A separate Licensee Event Report need not be submitted if the Safety Limit Violation Report meets the requirements of 10 CFR 50.73 (b) in addition to the requirements above.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November, 1972.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. Systems Integrity Program implementation.

h. Iodine Monitoring Program implementation.

i. PROCESS CONTROL PROGRAM implementation.

j. OFFSITE DOSE CALCULATION MANUAL implementation.

k. Quality Assurance Program for effluent and environmental monitoring.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License.

c. The change is documented and subsequently reviewed and approved within 14 days of implementation, in accordance with the requirements of Specification 6.8.2.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement unless otherwise noted.

STARTUP REPORTS

6.9.1.1 A summary report of plant startup and power escalation testing will be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details requested in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events, (i.e., initial criticality, completion of startup test program, and the resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL AND SEMIANNUAL REPORTS

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr. and their associated man-rem exposure according to work and job functions[1] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources should be assigned to specific major work functions.

b. A list of the reactor vessel material surveillance capsules installed in the reactor at the end of the report period and a summary of any withdrawals or insertions of capsules during the report period. In supplying this information, the ownership of each capsule shall be indicated and the irradiation location in the vessel of each capsule which was inserted during the report period shall be identified.

c. A routine Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

The Annual Radiological Environmental Operating Report shall include summaries, interpretations, unachievable LLDs, and an analysis of trends of the results of the radiological environmental studies and previous Annual Radiological Environmental Operating Reports and an assessment of any observed impacts of the plant operation on the environment. If harmful effects or

[1]This tabulation supplements the requirements of 20.407 of 10 CFR Part 20.

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Management and Program Analysis, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office, submitted no later than the 15th of each month following the calendar month covered by the report.

DELETED

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement, Region II, within the time period specified for each report. These reports shall be submitted covering the activities identified below. A separate Licensee Event Report, when required by 10 CFR 50.73 (a), need not be submitted if the Special Report meets the requirements of 10 CFR 50.73 (b) in addition to the requirements of the applicable referenced Specification.

a. ECCS Actuation, Specification 3.5.2 and 3.5.3.

b. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

c. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.4.

d. Seismic event analysis, Specification 4.3.3.3.2.

e. Inoperable Fire Detection Monitoring Instrumentation, Specification 3.3.3.7.

f. Specific Activity, Specification 3.4.8.

g. Results of Steam Generator Tube Inspection. Specification 4.4.5.5.b.

h. Inoperable Fire Suppression System, Specification 3.7.11.1., 3.7.11.2, 3.7.11.3, and 3.7.11.4.

i. Dose due to radioactive materials in liquid effluents in excess of specified limits, Specification 3.11.1.2.

j. Dose due to noble gas in gaseous effluents in excess of specified limits, Specification 3.11.2.2.

k. Total calculated dose due to release of radioactive effluents exceeding twice the limits of Specifications 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b., 3.11.2.3.a, or 3.11.2.3.b (required by Specification 3.11.3).

l. Dose due to Iodine-131, Tritium, and radioactive particulates with greater than eight day half-lives, in gaseous effluents in excess of specified limits, Specification 3.11.2.3.

m. Failure to process liquid radwaste, in excess of limits, prior to release, Specification 3.7.13.2.

n. Failure to process gaseous radwaste, in excess of limits, prior to release, Specification 3.7.13.3.

o. Measured levels of radioactivity in environmental sampling medium in excess of the reporting levels of Table 3.12-2, when averaged over any quarterly sampling period, Specification 3.12.1.1.

SPECIAL REPORTS(Cont'd.)

p. Inoperable Mid or High Range Noble Gas Effluent Monitoring Instrumentation, Specification 3.3.3.9.

q. Inoperable explosive gas monitoring instrumentation, Specification 3.3.3.10.

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time intervals at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed source and fission detector leak tests and results.

i. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components identified in Table 5.7.-1.

g. Records of training and qualification for current members of the plant staff.

h. Records of inservice inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the PRC and NGRC.

l. Records for Environmental Qualification which are covered under the provisions of paragraph 6.13.

m. Records of analytical results required by the Operational Radiological Environmental Monitoring Program.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c) (2) of 10 CFR 20:

a. A High Radiation Area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 90 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 AND BACKGROUND

The Nuclear Regulatory Commission (NRC), on August 29, 1983, in Federal Register Vol. 48, No. 168, amended its regulations, 10 CFR 50.72 and 50.73, to revise immediate reporting requirements and to establish a new reporting system for significant events at nuclear power plants. On December 19, 1983, the NRC issued Generic Letter No. 83-43, Reporting Requirements of 10 CFR 50, Sections 50.72 and 50.73, and Standard Technical Specifications (STSs), informing all licensees of the revision to Section 50.72, Immediate Notification Requirements, and the addition of a new Section 50.73, Licensee Event Report System. The licensees were requested to update their Technical Specifications (TSs) to include the new requirements, and model TSs were provided showing the revisions that should be made in the "Administrative Controls" and "Definitions" sections. The letter also requested the licensees to review and update other areas of the TSs concerning reportability, as required.

On April 23, 1984, Florida Power Corporation (the licensee) submitted a proposed amendment to Facility Operating License No. DPR-72 for the Crystal River Unit 3 Nuclear Generating Plant. The proposed amendment contained changes to the facility TSs in response to Generic Letter 83-43.

EVALUATION

The proposed changes to the TSs submitted by the licensee in response to Generic Letter 83-43 were patterned after the model TSs provided in the Generic Letter. In addition to the changes in the Administrative Controls and Definitions sections, modifications in other sections were necessary for consistency with the new regulations.

The immediate reporting requirements of 10 CFR 50.72 were not cited in the TSs since these regulations stand by themselves and inclusion in the TSs would be redundant. The new Licensee Event Report (LER) system of 10 CFR 50.73 was incorporated by providing the definition of the new term "REPORTABLE EVENT" and deleting the term "REPORTABLE OCCURRENCE". The new regulations are incorporated in Section 6.6 (Reportable Events). Sections 6.9.1.7 (Reportable Occurrences), 6.9.1.8 (Prompt Notification with Written Followup), and 6.9.1.9 (Thirty Day Written Reports) are deleted. Also, references to the deleted sections were modified in the TSs for consistency.

In addition to the changes suggested by the Generic Letter, there are several other necessary reporting requirement revisions due to the revised 10 CFR 50.73. Specifically, the statement "...in lieu of any other report required by Specification 6.9.1..." was deleted from Specifications 3.3.3.7, 3.7.11.1, 3.7.11.2, 3.7.11.3, and 3.7.11.4. Additionally, the requirement in Specification 3.4.8 to prepare and submit a REPORTABLE OCCURRENCE was replaced by a Special Report. Reports on abnormal degradation of the containment structure referenced in Surveillance 6.1.6.5 would be reported pursuant to 10 CFR 50.73, not Specification 6.9.1.

Finally, Section 6 was modified to allow LER requirements to be followed when a report must be submitted to the Commission that could also be required by 10 CFR 50.73. This eliminates the unnecessary time and paperwork of an additional report.

We have reviewed the above changes and find that they are administrative in nature, consistent with the guidance in Generic Letter 83-43, have no effect on safety, and conform with STSs. Therefore, the changes are acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment relates to changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need 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: July 17, 1986

Principal Contributor: B. Mozafari