Florida Power
CORPORATION
Crystal River Unit 3
Docket No. 50-302

July 7, 1997
3F0797-08

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555-0001

Subject: Reply to Notice of Violation and Exercise of Enforcement Discretion
(NRC Inspection Report No. 50-302/97-06)
NRC to FPC letter, 3N0697-04, dated June 5, 1997

Dear Sir:

In the subject letter, Florida Power Corporation (FPC) received a Notice of Violation. This correspondence provides our response to the Violation.

Sincerely,

Roy A. Anderson
Senior Vice President
Nuclear Operations

Attachments

RAA/TWC

cc: Regional Administrator, Region II
NRR Project Manager
Senior Resident Inspector

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IE01

STATE OF FLORIDA

COUNTY OF CITRUS

Roy A. Anderson states that he is the Senior Vice President, Nuclear Operations for Florida Power Corporation; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information, and belief.

Roy A. Anderson
Senior Vice President
Nuclear Operations

Sworn to and subscribed before me this 7th day of July, 1997, by Roy A. Anderson, who is personally known to me.

Pamela M. Layton

Signature of Notary Public
State of Florida

Pamela M Layton

Stamp Commissioned Name of Notary Public

Pamela M. Layton
MY COMMISSION # CC512982 EXPIRES
November 29, 1999
BONDED THRU TROY FAIN INSURANCE, INC.

ATTACHMENT 1
FLORIDA POWER CORPORATION
NRC INSPECTION REPORT NO. 50-302/97-06
REPLY TO A NOTICE OF VIOLATION

VIOLATION 50-302/97-06

10 CFR 50.59(a)(1) states, in part, that licensees may make changes to the facility or procedures as described in the safety analysis report, without prior Commission approval, unless the proposed change involves an unreviewed safety question (USQ). 10 CFR 50.59(a)(2) states, in part, that a proposed change shall be deemed to involve a USQ (i) if the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased or (ii) if a possibility for a malfunction of a different type than any evaluated previously in the safety analysis report may be created. 10 CFR 50.59(b)(1), in part, states that a licensee shall maintain records of changes in the facility and of changes in procedures made pursuant to this section. These records must include a written safety evaluation which provides the bases for the determination that the change does not involve a USQ. 10 CFR 50.59(c) states that a licensee who desires to make a change in the facility or procedures described in the safety analysis report which involves a USQ shall submit an application for amendment of his license pursuant to 10 CFR 50.90.

Prior to May 2, 1996, the facility Final Safety Analysis Report (FSAR) described the facility's mitigation strategy for a design basis small break Loss of Coolant Accident (LOCA), and that strategy included two operator actions. These actions were: initiate high pressure injection (HPI) flow through all four injection lines within 20 minutes (per FSAR Table 6-19); and balance flows in the HPI injection lines within 20 minutes (per FSAR Sections 6.1.3.1.2 and 4.2.2.5.7.2). The NRC had previously (in 1979) approved use of one operator action to mitigate a design basis small break LOCA, i.e., initiate HPI flow through all four injection lines by 10 minutes after the LOCA.

Contrary to the above, on the dates indicated below, the licensee made changes to the facility and procedures described in the FSAR that involved USQs. The changes involved the addition of the operator actions described below to ensure that the design basis requirements for small break LOCA mitigation were met. The FSAR itself was also changed to include some of the operator actions. These changes were made based on inadequate safety evaluations, and as a result, a license amendment was not sought for conditions that involved USQs.

The facility was changed by analysis in Calculation M96-0032, Reevaluation of HPI Requirements During Small Break Loss of Coolant Accidents, dated May 2, 1996 such that additional operator actions were required to mitigate the consequences of a design basis small break LOCA. However, the additional operator actions had not been approved by the NRC to be relied on for mitigation of a design basis small break LOCA. The operator actions added or changed included:

(1) isolate letdown within 10 minutes.
(2) isolate normal makeup within 20 minutes.
(3) isolate reactor coolant pump (RCP) seal injection within 20 minutes.
(4) isolate a broken HPI injection line within 20 minutes, and
(5) control steam generator level above the Emergency Feedwater Initiation and Control automatic setpoint within 20 minutes.

In summary, this change added operator actions [(1), (2), (3), and (5) above] and changed one operator action [(4) above, which replaced the previous operator action to balance flows in the HPI injection lines within 20 minutes] in the facility's mitigation strategy for a design basis small break LOCA.

Procedures described in the FSAR, i.e., the emergency operating procedures, were changed by Short Term Instructions (STI) 95-0061, effective November 8, 1995 to February 8, 1996; STI 96-0068, effective February 8, 1996 to May 6, 1996; and Revision 4 to Emergency Operating Procedure EOP-03, dated May 2, 1996; to add the operator action to isolate the RCP seal injection. The remaining operator actions had been in the emergency operating procedures since the late 1970s, but at least four of the them had not been relied upon to satisfy the design basis as stated in the FSAR.

The FSAR was changed by Revision 23, titled "FSAR Revision due to HPI Reevaluation," dated November 18, 1996, to incorporate the results of Calculation M96-0032 into the FSAR. The change included the operator actions listed above, with the exception of action (5). The safety evaluation for FSAR Revision 23 was dated April 30, 1996.

The required safety evaluations that supported Revision 23 to the FSAR, STI 95-0061, STI 96-0068, and Emergency Operating Procedure EOP-03, Rev. 4, were inadequate in that they failed to recognize the introduction of USQs. (There was no separate safety evaluation for Calculation M96-0032.) The inadequacies involved a failure to recognize that the increase in the number of operator actions required to mitigate a design basis small break LOCA introduced the possibility of a malfunction of a different type than any evaluated previously in the FSAR, and also increased the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the FSAR. The changes introduced additional opportunities for operator errors. The inadequacies also involved a failure to recognize that addition of the action to isolate RCP seal injection increased the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the FSAR in that it increased the probability of seal failure. Therefore, the changes involved USQs.

The safety evaluation for FSAR Revision 23 was also inadequate in that it failed to address, and failed to ensure that the FSAR included all of the required operator actions for small break LOCA mitigation that were stated in Calculation M96-0032. Operator action (5) was not addressed by the safety evaluation for FSAR Revision 23 and was not included in the FSAR revision.

ADMISSION OR DENIAL OF THE ALLEGED VIOLATION

Florida Power Corporation (FPC) accepts the violation. The following Background and Specific Operator Actions sections contain references to prior docketed correspondence indicating where the NRC has previously reviewed the subject operator actions and is provided to assist NRC staff in its evaluation.

BACKGROUND

As discussed in NRC Special Inspection Report 97-06, the emergency core cooling system (ECCS) and other plant systems were originally designed to operate automatically for the first 20 minutes of a design basis event with a limited number of operator actions. In the late 1970s, small break LOCA was generically identified as not being enveloped by large break LOCAs. In a letter dated September 26, 1978, NRC provided FPC with the staff position regarding allowable operator actions for which credit may be taken following a Condition III event (small LOCA). This position defined "simple" and "complex" operator actions.

Based on subsequent small break LOCA analyses (cold leg pump discharge break, HPI line break, HPI pinch break) it has been determined that during certain small break LOCA scenarios, operator actions may be required to ensure depressurization and cooldown of the reactor coolant system (RCS). For example, subcooling margin may be inadequate prior to automatic engineered safeguards (ES) actuation requiring operator action to ensure HPI is providing injection of water to the core. The loss of subcooling margin is described in the B&W Abnormal Transient Operating Guidelines (ATOG) as a symptom of a small break LOCA. The B&W ATOG was reviewed by the NRC through Generic Letter 83-31, "Safety Evaluation of Abnormal Transient Operating Guidelines." B&W Owners Group (BWOG) Letter to NRC dated September 11, 1985 from M. A. Linn to G. C. Vissing submitted a copy of the BWOG Emergency Operating Procedures Technical Bases Document (TBD) which consolidated the ATOG technical bases in addressing comments contained within Generic Letter 83-31. The CR-3 plant specific ATOG was submitted to NRC by FPC letter dated March 25, 1983 as part of its Procedure Generation Package (PGP) submittal. Subsequently, FPC received a Safety Evaluation Report of its PGP by NRC letter dated April 6, 1990. FPC's ATOG/TBD is based on the B&W Owners Group ATOG/TBD.

FPC's discovery of errors in the small break LOCA analyses and related issues, was reported in LERs 95-026, 96-006, and 96-007. FPC also provided the NRC in a letter dated May 22, 1996, the results of new small break LOCA analyses in support of efforts related to improved HPI instrumentation indicating a change in peak clad temperature. This letter also states that previous small break LOCA analyses did not bound the actual HPI flow splits resulting in operator actions required to mitigate the event.

SPECIFIC OPERATOR ACTIONS

(1) Isolate letdown within 10 minutes

Isolating letdown requires closing one valve (MUV-49) from the control room. This action is a simple action as defined by NRC guidance described in letter dated September 26, 1978. It was part of the original accident analysis and was contained in EP-106,"Loss of RC/RC System Pressure," Revision 1, dated January, 1974. MUV-49 is designed to close on an ES system actuation signal.

Isolating letdown was recognized as a required operator action in the B&W ATOG document. NRC to FPC letter dated July 6, 1979 provided a safety evaluation report (SER) for action taken in response to a May, 1979 Commission Order. With respect to Section IV.1.d of that order, to complete small break LOCA analyses and implement operator actions, the SER recognized that CR-3 revised Emergency Procedures EP-106 and EP-103, "Loss of RC Flow/RC Pump Trip," which define operator actions in response to a spectrum of break sizes. The SER states "The procedure [EP-106] was judged to provide adequate guidance to the operators to cope with a small break LOCA."

The small break LOCA analysis provided by Framatome Technologies, Inc. (FTI) in May, 1996 indicated that during a small break LOCA, the RCS pressure may not immediately reach the ES actuation setpoint and therefore, the operator is required to close MUV-49 manually. This action meets the criteria for "simple" operator action (pushing a button or turning a switch).

(2) Isolate normal makeup within 20 minutes

The method of isolating normal makeup is to close a single valve. This action was introduced into the EOPs (EP-106 Revised June, 1978) based on a B&W small break LOCA analysis performed in April, 1978. In order to isolate a broken line for an HPI line break, normal makeup must be isolated. By isolating normal makeup, HPI flow can be measured more accurately. As indicated above, the SER stated EP-106 provided adequate guidance to the operators to cope with a small break LOCA. However, FPC letters to NRC dated February 28 and April 5, 1979 stated that normal makeup did not need to be isolated. Therefore, FPC agrees that this operator action, when introduced into the revised small break LOCA analysis in 1996, should have been re-submitted for NRC review.

(3) Isolate reactor coolant pump (RCP) seal injection within 20 minutes

The method of isolating RCP seal injection is to close a single valve. This action was introduced into the EOPs in 1996. In answering a request for additional information related to resolution of small break LOCA problems, FPC provided a letter to NRC dated February 28, 1979 stating that isolation of RCP seal injection was not needed based on results of a flow analysis indicating HPI flow was satisfactory with seal injection not isolated. The need to isolate RCP Seal Injection was subsequently determined to be necessary based on the fact that operators relied on non-Reg Guide 1.97 instrumentation to measure this flow when determining HPI pump runout flow limits. This was reported as a design basis issue in LER 95-026-00 dated December 7, 1995.

In addition, during Refuel 10, FPC discovered that worst case instrument error may result in inadequate HPI flow, also necessitating isolation of RCP seal injection. This was reported as a design basis issue in LER 96-006-00 dated February 29, 1996.

FPC agrees this operator action should have been submitted to NRC for review and also agrees that the action was improperly evaluated with respect to the possibility of seal damage. Further review of the vendor technical manual for the RCP seals indicates that, for an idle pump, RCP seal controlled bleed-off (CBO) valves are required to be closed after 90 seconds if seal injection has not been restored. This action restricts the heatup rate of the seal cartridge to minimize the possibility of seal damage.

(4) Isolate a broken HPI line within 20 minutes

The high flow line condition is indicative of an HPI line break. The HPI line break small break LOCA results in a limited amount of HPI flow going to the core as a result of nominal backpressure against the broken line.

Operator action to Isolate RCS leaks was required as part of the mitigation strategy for maximizing HPI flow to the core in the CR-3 ATOG.

FPC issued LER 96-007 on March 15, 1996 to report a design basis condition involving HPI flow instrumentation. The flow deficiencies described therein were addressed by revised small break LOCA analyses provided by Framatome Technologies, Incorporated (FTI) in April 1996, which required isolation of the affected HPI line instead of balancing. The most recent FTI analyses have provided new isolation criteria.

While the balancing criteria did change to isolation criteria, the purpose of this operator action, of maximizing HPI flow to the core, did not change. However, FPC agrees this action should have been submitted for review.

(5) Control steam generator level above the Emergency Feedwater Initiation and Control automatic setpoint within 20 minutes

This step was added to the emergency procedures in December 1979 to manually raise steam generator levels to 95%. NRC letter to FPC dated July 6, 1979 provides a SER for actions taken in response to Commission Order dated May 16, 1979 and states, "A principal finding of our generic review [of B&W analyses entitled 'Evaluation of Transient Behavior and Small RCS Breaks in the 177 Fuel Assembly Plant'] is a reconfirmation that LOCA analyses of breaks at the lower end of the

...spectrum...demonstrate that a combination of heat removal by the steam generators and the HPI system combined with operator action ensure adequate core cooling...These results are applicable to CR-3 considering the ability to manually start the redundant EFW pumps and HPI pumps from the control room, assuming failure of automatic EFW actuation."

NRC letter to FPC dated March 8, 1983 provided a summary of a meeting held February 23, 1983 with B&W concerning small break LOCA procedures and maintaining proper steam generator water level. It was determined that raising the level in the steam generators to 95% of the operating range would assure natural circulation if the RCS was saturated. NRC letter to FPC dated August 30, 1985 provides a SER for NUREG 0737 Item II.K.3.30, "Small Break LOCA Methods." Section III.5.a of the SER states "the timing of operator action to raise the secondary system water level to 95% was found not to be critical."

In September 1992, a change to Abnormal Procedure AP-380, "Engineered Safeguards Actuation," revised steam generator levels required for inadequate subcooling margin to provide a "band" for control to encompass both the automatic setpoint and an achievable operator control range. The control band of 80-90% was determined to be acceptable when considering analytical values and instrument error. Reference was made to the Technical Basis Document for the EOPs to support the change. In May 1996, when the emergency operating procedure was revised to include the small break LOCA analysis assumed operator action to raise steam generator levels to 95%, FPC failed to recognize the need to submit this action for NRC review.

REASON FOR THE VIOLATION

Florida Power Corporation erroneously believed that NRC review of operator actions related to symptom-oriented conditions was sufficient via NRC's acceptance of its plant specific technical guidelines (PSTG) per NUREG 1358 and NRC Inspection Procedure 42001, "Emergency Operating Procedures." Also, FPC's guidance was not consistent with NRC's interpretation of the regulation as identified in NRC memo of July 30, 1996 in response to Technical Assistance Request 95-013 which addressed issues regarding St. Lucie diesel generator fuel oil transfer system leak isolation using operator action in place of automatic action.

A contributing cause of this violation was a lack of a formal program which provided for training and qualification of personnel involved with the 10 CFR 50.59 process. As a result, FPC had not maintained a consistent level of quality in preparation and reviews of 10 CFR 50.59 evaluations. Further, based on the lessons learned from the Millstone core offloading event in 1995 as reflected in Interim Part 9900 Inspection Guidance, FPC took limited action to implement corresponding changes to its 10 CFR 50.59 reviews.

CORRECTIVE STEPS THAT HAVE BEEN TAKEN AND THE RESULTS ACHIEVED

FPC has upgraded its 10 CFR 50.59 review process effective March 31, 1997, establishing a "stand-alone" safety evaluation format that requires an integrated discussion of the proposed change and its effects.

Training to enhanced 10 CFR 50.59 guidance has been provided to selected individuals who are identified as qualified to write and review safety evaluations (FPC Restart Issue OP-5). The scheduled training for the upgraded process was completed in mid June; however, classes continue on an as-needed basis to ensure new employees and other personnel are trained as necessary. This training emphasizes the quality and thoroughness of the 10 CFR 50.59 evaluation.

FPC established a Safety Analysis Group (SAG) which is currently staffed with FPC engineering and contractor personnel who are knowledgeable in design basis accident analyses and the 10 CFR 50.59 process. The SAG organization is tasked with reviewing 10 CFR 50.59 evaluations for EOP changes as well as modifications and selected procedure changes. They are responsible for the CR-3 PSA Model, safety analysis, and fuel management.

FPC submitted Technical Specification Change Request (TSCRN) 210 on June 14, 1997 with proposed license amendments to support operation with hardware changes primarily involving the Emergency Feedwater (EFW), High Pressure Injection, Emergency Feedwater Initiation and Control Systems, and the Emergency Diesel Generators (EDGs), as well as associated licensing and design basis changes. TSCRN 210 includes a list of all operator actions required to mitigate the consequences in the first 20 minutes of certain small break LOCA scenarios with concurrent loss of offsite power.

TSCRN 210 reflects the addition of one simple manual operator action which replaces several complex manual actions. The EOPs will be revised prior to restart of the unit to require the operator, upon loss of subcooling margin, if engineered safeguards (ES) has not actuated, to initiate manual HPI and Reactor Building Isolation and Cooling (RBIC). This simple operator action automatically isolates letdown, initiates HPI flow, isolates normal makeup (contingency actions are provided within 20 minutes if power is not available), isolates RCP seal control bleed off valves, actuates EFIC, and initiates emergency reactor building cooling.

CORRECTIVE STEPS THAT HAVE OR WILL BE TAKEN TO AVOID FURTHER VIOLATIONS

FPC will develop further 10 CFR 50.59 review guidance regarding manual operator actions and provide training by July 29, 1997. FPC intends to use approved, published NRC guidance to supplement its 10 CFR 50.59 program guidance. The guidance will ensure that changes or additions to manual operator actions assumed in the design basis analyses are evaluated with respect to replacement of automatic actions with manual actions.

As part of its response to a request for public comments of recently published NUREG 1606, "Proposed Regulatory Guidance Related to 10 CFR 50.59," FPC will comment as to how 10 CFR 50.59 is intended or expected to apply to the various categories of operator actions contained within EOPs. These may include: operator actions assumed in design basis accidents previously reviewed by NRC as reflected in event-based accident analyses; those actions specified by symptom oriented conditions as reflected in NRC-approved technical basis documents which form a part of the PGP; those actions required by new requirements (such as 10 CFR 50 Appendix R); those actions beyond the design basis such as may be required in severe accident management; contingency operator actions, and actions that must be taken after the first 20 minutes of an accident.

DATE WHEN FULL COMPLIANCE WILL BE ACHIEVED

FPC will be in full compliance on July 29, 1997 upon issuance of enhanced guidance and completion of training regarding proposed changes to manual operator actions.

ATTACHMENT 2

The following table contains a listing of commitments contained in this response:

Response Section	Commitment	Due Date
Page 9	The EOPs will be revised to require the operator, upon loss of subcooling margin, if engineered safeguards (ES) has not actuated, to initiate manual HPI and Reactor Building Isolation and Cooling (RBIC) pushbuttons.	Prior to Restart
Page 10	FPC will develop further guidance regarding manual operator actions using approved, published NRC guidance to supplement its 10 CFR 50.59 program guidance. Training will also be provided.	July 29, 1997
Page 10	As part of its response to a request for public comments of recently published NUREG 1606, "Proposed Regulatory Guidance Related to 10 CFR 50.59," FPC will comment as to how 10 CFR 50.59 is intended or expected to apply to the various categories of operator actions contained within EOPs.	July 7, 1997