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

June 16, 1997
3F0697-12

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

Subject: NRC Notice of Violation, Integrated Inspection Report No.50-302/96-19,
NRC to FPC letter, 3N0597-13, dated May 16, 1997

Dear Sir,

In the subject letter, Florida Power Corporation (FPC) received a request for a supplemental response to the subject Notice of Violation. This correspondence provides the requested supplement to our previous response.

Sincerely,

John Paul Cowan
Vice President
Nuclear Production

JPC/JPB

Attachments

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

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IE01

Attachment 1
FLORIDA POWER CORPORATION
NRC INSPECTION REPORT NO. 50-302/96-12 & 96-19
REPLY TO A NOTICE OF VIOLATION

VIOLATION A.

10 CFR 50.59, "Changes, Tests and Experiments," provides, in part, that the licensee may make changes in the facility or procedures as described in the safety analysis report (SAR) without prior Commission approval, unless the proposed change involves a change in the Technical Specifications (TS) or an unreviewed safety question (USQ). A proposed change shall be deemed to involve a USQ if the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR may be increased, if a possibility for an accident or malfunction of a different type than any evaluated previously in the SAR may be created, or if the margin of safety as defined in the basis for any TS is reduced. 10 CFR 50.59 further requires that a written safety evaluation be documented providing the bases for a determination that the changes do not involve a USQ.

The TS bases for TS 3.8.1, AC Sources - Operating, states that the service rating of the emergency diesel generator (EDG) is, in part, 3251 to 3500 kilowatts (KW) on a cumulative 30 minute basis.

The Final Safety Analysis Report (FSAR), Rev. 19, dated December 21, 1994, Section 8.2.3, Sources of Auxiliary Power, provides the load ratings for both EDGs, including a 2851 - 3000 KW cumulative 2000 hour rating and a 3251 - 3500 KW cumulative 30 minute rating. (The maximum load rating shown for any period of time is 3500 KW). It also states that the "A" EDG auto-connected load is within the 2000 hour rating at one minute into the scenario. FSAR, Rev. 20, dated April 1, 1994, Table 8-1, Emergency Diesel Generator "A" Auto & Manually Connected Loads, lists the largest auto-connected load as make-up pump 1A (615.5 KW). This FSAR information remained current through 1996.

The FSAR, Rev. 10, dated July 1, 1988, Table 8-1, Emergency Diesel Generator "A" Auto & Manually Connected Loads, lists the largest auto-connected load as make-up pump 1A (615.5 KW). This FSAR information remained current through 1990.

The FSAR, Rev. 10, dated July 1, 1988, Section 10.5, Emergency Feedwater (EFW) System, states that upstream of the turbine-driven emergency EFW pump turbine steam supply line, there are redundant, normally closed direct current (DC) motor operated valves (ASV-5 and ASV-204) which are opened upon actuation from the emergency feedwater initiation and control (EFIC) system. FSAR, Rev. 8, dated July 1, 1987, Section 7.2.4, Emergency Feedwater Initiation and Control, states that the EFIC trip module located in the "A" cabinet actuates the "A" train of EFW (motor-driven pump) and the trip module located in the "B" cabinet actuates the "B" train of EFW (turbine-driven pump) (EFP-2). This FSAR information was the first description of the EFIC system, and it remained current through 1992. Section 7.2.4 of the FSAR was revised on January 17, 1993, Rev. 18, as follows: "The trip module located in the "A" cabinet starts the "A" train motor-driven EFW pump and the "B" train turbine-driven EFW pump. The trip module located in the "B" cabinet starts only the "B" train turbine-driven EFW pump. The starting of both EFW pumps on "A" train EFIC actuation is necessary to assure that the turbine-driven pump will be operable in the event of a failure of the ES "B" 250/125V DC system coincident with a loss of offsite power and a [engineered safeguards] actuation. Under this scenario, EFP-2 will be relied upon to share the emergency feedwater load with the motor driven emergency feedwater pump in order to decrease the electrical load on diesel generator EDG-3A." This FSAR information remained current through 1996.

1. Contrary to the above, in April 1996, the licensee made a change to the facility as described in the FSAR, which involved three USQs, without prior Commission approval. Specifically, the modification, installed by Modification Approval Record (MAR) 96-04-12-01 changed the EFW initiation logic to allow the motor-driven EFW pump to provide all EFW during certain analyzed accidents which increased the calculated post-accident motor-driven EFW pump load from about 616 KW to about 666 KW. As a result, the "A" EDG accident loads were in excess of the limits specified in FSAR Section 8.2.3, TS 3.8.1 Basis (3500 KW limit), TS surveillance requirement (SR) 3.8.1.11 Basis (3100 KW one-minute load), and TS SR 3.8.1.8 Basis (616 KW largest single post-accident load that could be rejected). This change reduced the margin of safety as defined in the FSAR and three TS Bases, resulting in three USQs. The 10 CFR 50.59 safety evaluation for this modification was inadequate in that it did not address electrical loading effects on the "A" EDG and did not recognize the USQs.

2. Contrary to the above, in April 1996, the licensee made a change to a procedure as described in the FSAR, which involved three USQs, without prior Commission approval. Specifically, Emergency Operating Procedure EOP-13, EOP Rules, was changed by Rev. 2 to require operators to take manual control of the motor-driven EFW pump to increase EFW flow under certain conditions, resulting in an increase in EFW pump load from about 666 KW to about 713 KW. As a result, the "A" EDG accident loads were in excess of the limits specified in the FSAR and TS 3.8.1 Basis (3500 KW limit), TS SR 3.8.1.11 Basis (3100 KW one-minute load), and TS SR 3.8.1.8 Basis (616 KW largest single post-accident load that could be rejected). This change reduced the margin of safety as defined in the FSAR and three TS Bases, resulting in three USQs. The 10 CFR 50.59 safety evaluation for this procedure change was inadequate in that it did not address electrical loading effects on the "A" EDG and did not recognize the USQs.

3. Contrary to the above, in June 1990, the licensee made a change to a procedure as described in the FSAR, which involved a USQ, without prior Commission approval. Specifically, Operating Procedure OP-402, Makeup and Purification System, was changed by Rev. 64 to allow operators to select, for Engineered Safeguards, the swing "B" makeup pump to either EDG. This resulted in an increase in the largest single post-accident load on the "A" EDG, from 616 KW ("A" makeup pump) to 691 KW ("B" makeup pump). The 691 KW exceeded the largest single post-accident load, that could be rejected by the "A" EDG, specified in the FSAR (616 KW) and in TS SR 3.8.1.2.2 (515 KW). This change in the largest single post-accident load required a TS change which was not made, and therefore resulted in a USQ. The 10 CFR 50.59 safety evaluation for this procedure change was inadequate in that it did not address electrical loading effects on the "A" EDG and did not recognize the USQ.

4. Contrary to the above, in May 1987 and in March 1992, the licensee made changes to the facility as described in the FSAR, which involved a USQ, without prior Commission approval. Specifically, modifications modification package T87-10-09-01 and modification package 87-10-09-01A changed the EFW system electrical power supply for the turbine-driven EFW pump alternate steam admission valve, ASV-204, from "B" train to "A" train DC power and changed the automatic opening of ASV-204 from a "B" train to an "A" train EFW initiation signal. The change introduced a USQ in that, in certain accident scenarios, a failure of the "B" battery would cause the turbine-driven EFW pump to go to runout with its flow control valves failed fully open which would increase the probability of failure of the turbine-driven EFW pump. If the event were also concurrent with a loss of offsite power, the "B" EDG would not operate (due to failure of the "B" battery). Also, the licensee's design basis relied on the turbine-driven EFW pump to share the EFW flow requirements with the motor-driven EFW pump in order to maintain the "A" EDG within its loading limits. The plant operated in various modes from 1987 through April 1996 with this design. The 10 CFR 50.59 safety evaluations for the modification package were inadequate in that they did not address hydraulics, potential net positive suction head (NPSH) problems, a resulting potential increase in the probability of a malfunction of the turbine-driven EFW pump, or consequential effects on the "A" EDG; and did not recognize the USQ.

5. Contrary to the above, in May 1996, the licensee made changes to the facility, which involved a USQ, without prior Commission approval. Specifically, the 10 CFR 50.59 safety evaluation for modification package 96-04-12-01 (installed in May 1996) was inadequate in that the safety evaluation did not identify that removal of the automatic open signal from valve ASV-204 increased the probability of occurrence of a malfunction of equipment important to safety and therefore was a USQ. Removal of the automatic open signal from valve ASV-204 disabled one of the two automatic steam supplies to EFP-2, which reduced the reliability and increased the probability of a failure of EFP-2.

6. Contrary to the above, in 1994, the FSAR was revised in Rev. 21, dated December 1, 1994, Section 4.3.10.1, Boron Dilution, to add information on the boron precipitation methods following a loss of coolant accident (LOCA), and the 10 CFR 50.59 evaluation was inadequate in demonstrating that a USQ did not exist. Specifically, after identifying deficiencies in the active methods (decay heat drop line and the pressurizer auxiliary spray line) used for boron precipitation control, the FSAR and Design Basis Documents were inappropriately changed to specify flow through gaps in the reactor vessel internals (a passive method) as the first and preferred method. This departed from the original licensing basis of the plant. Also, flow through reactor vessel internal gaps had been identified as acceptable to the NRC (letter dated March 9, 1993) only as a backup method and not as the primary method.

These violations represent a Severity Level II problem (Supplement I).

<u>ADMISSION OR DENIAL OF THE ALLEGED VIOLATIONS</u>

FPC accepts these violations as described.

REASONS FOR THE VIOLATIONS

The violations resulted from shortcomings in the areas of engineering performance, configuration management, and regulatory performance. As discussed at the enforcement conference, the root causes of the violations cited under "A" are bounded by the causes that gave rise to the Management Corrective Action Program, Phase II (MCAP II). MCAP II has been shown through many NRC/FPC public meetings and docketed correspondence to contain comprehensive corrective actions that address the programmatic areas of concern associated with these violations.

The above violations resulted from deficiencies in the implementation of the 10 CFR 50.59 process and insufficient maintenance of the Design Basis Documents, such as the FSAR. The formal training and qualification program for personnel involved with the 10 CFR 50.59 process was not sufficient to ensure consistency in the performance of 10 CFR 50.59 evaluations. Achievement of the required level of consistency in 10 CFR 50.59 evaluations was limited due to the lack of a thorough understanding of the design basis.

CORRECTIVE ACTIONS THAT HAVE BEEN TAKEN AND THE RESULTS ACHIEVED

GENERAL CORRECTIVE ACTIONS

The plant was not returned to service following a shutdown in September 1996 for a turbine generator lube oil leak, because of the discovery of Unreviewed Safety Questions (USQs) and insufficient design margin as identified in FPC's October 28, 1996 letter to the NRC. As stated in correspondence to and from the NRC, the plant will not be returned to service until the issues identified in the October 28, 1996 letter are satisfactorily resolved.

FPC has taken a number of actions to improve its 10 CFR 50.59 process including:

1) Management conveyed expectations for effective 10 CFR 50.59 evaluations to those individuals involved with the process. FPC established and clarified management expectations to ensure the 10 CFR 50.59 Safety Evaluation is a stand alone document and not just a component of the change package. Immediate training was given on the importance and interpretation of 10 CFR 50.59. In particular, it was emphasized that the quality and thoroughness of the 10 CFR 50.59 evaluation was as important as the design change itself.

2) The Safety Analysis Group was established to perform independent reviews. This independent review assignment includes review of Safety Assessments (SA) and USQ Determinations (USQDs), i.e., 10 CFR 50.59 evaluations. This review group provides guidance, preliminary reviews at the conceptual and intermediate stages, and formal concurrence with final SA/USQDs. The group's initial focus was modification activities, but has been expanded to include review of SA/USQDs associated with selected procedure changes. This group will review safety assessments in a graded manner. A sample of changes not reviewed by the group will be assessed to assure expectations are being met without reliance on the independent review.

3) The 10 CFR 50.59 process procedure was revised effective April 1, 1997. Rigorous training of evaluators and reviewers, which is still in progress, has already resulted in comprehensive and detailed 10 CFR 50.59 evaluations. In addition, the guidance for conducting SA/USQDs has been centralized with all organizational units using the same procedure for guidance, training, and format.

4) Training/Qualification requirements have been established for personnel performing and reviewing 10 CFR 50.59 evaluations. Personnel training began in March 1997 and will continue until sufficient personnel are certified. Requalification training, as needed, will be an ongoing program. Only personnel who have been trained and qualified under the new program will be authorized to perform or review 10 CFR 50.59 evaluations after June 30, 1997. In the interim, the Safety Analysis Group is reviewing 10 CFR 50.59 evaluations to ensure high quality.

5) Modifications are currently required to have a complete 10 CFR 50.59 evaluation even if the established screening requirements would result in none required. Under limited conditions, the Design Engineering Manager may waive the requirement for this 10 CFR 50.59 safety evaluation if it is clear none is required. This process will continue until formal training and establishment of qualified reviewers for 10 CFR 50.59 evaluation is complete.

6) Interdisciplinary and interdepartmental reviews have been established through use of Design Review Boards (DRBs). Depending on the complexity of the modification, formal, structured and extensive conceptual, final and other design reviews are regularly performed. Third party independent design reviews have been and will continue to be obtained for a number of significant design changes.

7) The Plant Review Committee (PRC) is chaired by a senior manager with operations and engineering experience who reports to the Director Nuclear Plant Operations. The PRC's representation and processes have been formalized and strengthened as a result of self-assessments, critical feedback, and benchmarking with other utilities. AI-300, Plant Review Committee Charter, has been revised and PRC members trained on the enhanced guidance. Expectations are delineated and protocol formalized. Alternates are limited to those persons who have sufficient experience for this oversight committee.

As a result of these initiatives, FPC has noted significant improvements in the quality and thoroughness of recent safety evaluations.

CORRECTIVE ACTIONS THAT WILL BE TAKEN TO AVOID FURTHER VIOLATIONS

GENERAL CORRECTIVE ACTIONS

In addition to the above corrective actions that have been implemented, FPC will provide training on the application, interpretation, and expectations of Improved Technical Specifications (ITS) and ITS BASES in support of the design process to the Nuclear Operations Engineering Staff as a continuing effort. This item is currently being tracked as MCAP II item number C-ID-III-1, with a targeted completion date of December 31, 1997.

Several of the violations stem from EDG capacity, loading, and load management issues and EFW requirements for small break loss of coolant accident (SBLOCA) mitigation. FPC has committed to resolve these issues, which will be complete prior to restart, including licensing submittals and receipt of NRC approvals for amendments to the Crystal River Unit 3 (CR-3) License and ITS amendments. The program will include modifications to resolve the individual USQs identified in the violations and restore adequate design margins. These include modifications such as uprating the EDG, installing cavitating venturis in the EFW system, making EFIC improvements, and completing the ASV-204 modification.

FPC is confident that the improved programs implementing our 10 CFR 50.59 process, our design control program, and our corrective action program including the associated training will enhance self-identification and evaluation of design deficiencies or potential USQs. Prior to plant restart, FPC is engaged in an extensive System Readiness Review Program to assure that safety-related systems are in compliance with the licensing and design basis of the CR-3 facility.

Below, FPC provides additional, more specific corrective actions addressing the individual violations cited.

VIOLATION A.1.

CORRECTIVE ACTIONS THAT HAVE BEEN TAKEN AND THE RESULTS ACHIEVED

The 10 CFR 50.59 process procedure was revised effective April 1, 1997. Rigorous training of evaluators and reviewers, which is still in progress, has already resulted in comprehensive and detailed 10 CFR 50.59 evaluations. In addition, the guidance for conducting SA/USQDs has been centralized with all organizational units using the same procedure for guidance, training and format.

Training/Qualification requirements have been established for personnel performing and reviewing 10 CFR 50.59 evaluations. Personnel training began in March 1997 and will continue until sufficient personnel are certified. Only personnel who have been trained and qualified under the new program will be authorized to perform or review 10 CFR 50.59 evaluations after June 30, 1997. In the interim, the Safety Analysis Group is reviewing 10 CFR 50.59 evaluations to ensure high quality.

Interdisciplinary and interdepartmental reviews have been established through use of Design Review Boards (DRBs). Depending on the complexity of the modification, formal, structured and extensive conceptual, final and other design reviews are regularly performed. Third party independent design reviews have been and will continue to be obtained for a number of significant design changes to assist the DRBs.

Modifications are currently required to have a complete 10 CFR 50.59 evaluation even if the established screening requirements would result in none required. Under limited conditions, the Design Engineering Manager may waive the requirement for this 10 CFR 50.59 safety evaluation if it is clear none is required.

CORRECTIVE ACTIONS THAT WILL BE TAKEN TO AVOID FURTHER VIOLATIONS

Procedural changes will be made to improve the definition of the responsibilities of the design engineer, verification engineer, and the EDG Load Management Program, and the verification of calculations. This enhanced procedural guidance will include 10 CFR 50.59 evaluations, engineering interdisciplinary interface, design input considerations, external interfacing for modifications and calculations, analysis/calculation review and verification, timely updating of engineering documents (design basis documents, computer data bases), design verification process, and engineering review prior to return to service of modifications. The procedural changes will be accomplished by September 15, 1997.

VIOLATION A.2.

CORRECTIVE ACTIONS THAT HAVE BEEN TAKEN AND THE RESULTS ACHIEVED

The 10 CFR 50.59 process procedure was revised effective April 1, 1997. Rigorous training of evaluators and reviewers, which is still in progress, has already resulted in comprehensive and detailed 10 CFR 50.59 evaluations. In addition, the guidance for conducting SA/USQDs has been centralized with all organizational units using the same procedure for guidance, training and format.

Training/Qualification requirements have been established for personnel performing and reviewing 10 CFR 50.59 evaluations. Personnel training began in March 1997 and will continue until sufficient personnel are certified. Only personnel who have been trained and qualified under the new program will be authorized to perform or review 10 CFR 50.59 evaluations after June 30, 1997. In the interim, the Safety Analysis Group is reviewing 10 CFR 50.59 evaluations to ensure high quality.

In addition, FPC has taken the following specific corrective actions to address Violation A.2.:

a) FPC has established stringent review requirements on Engineering calculations. All formal Engineering calculations require verification, including "case studies." In addition, formal Engineering calculations require reviews from the Operations Department.

b) Personnel involved in the program were counseled to ensure thorough understanding of the errors made.

The result of these corrective actions is that the engineering calculation review process has improved. This review process identified the manual EFW flow control issue associated with this violation.

CORRECTIVE ACTIONS THAT WILL BE TAKEN TO AVOID FURTHER VIOLATIONS

In addition to the corrective actions discussed in the General Corrective Actions section, FPC will take the following specific corrective actions to address Violation A.2.:

a) The Final Safety Analysis Report (FSAR), Enhanced Design Basis Document (EDBD), and Analysis Basis Document (ABD), will be updated to show EDG and EFW equipment dependencies and equipment limitations. The update will be completed by November 20, 1997.

b) Design basis training will be provided to the following Operations personnel: Nuclear Shift Supervisor, Assistant Nuclear Shift Supervisor, Nuclear Shift Manager and Operations Engineering. The training will include a discussion of the Design Basis and Design Basis Accidents, initiating events, passive failures, active failures train dependencies, equipment operation/availability assumptions in the analysis of the various accidents, and what plant components are required for the operation of the required systems. Initial training will be completed by December 31, 1997 and will be continuing in nature.

c) The EDGs are being uprated to establish design margin over the current loading. License amendments will be submitted to revise the design and licensing basis and the ITS BASES to reflect this uprate prior to restart from the present shutdown.

d) The emergency operating procedures (EOPs) are being revised as part of the EOP Enhancement Program. The revised procedures will have appropriate reviews by design personnel and will be issued prior to restart from the present outage.

   Administrative instruction AI-400F (New Procedures and Procedure Change Processes For EOPs, APs, and Supporting Documents) dated 03/31/97, provides specific requirements for ensuring proper reviews of EOPs and APs during the revision cycle, including the requirement for reviews by both design engineering and system engineering.

   These corrective actions provide reasonable assurance that similar violations will not recur.

VIOLATION A.3.

CORRECTIVE ACTIONS THAT HAVE BEEN TAKEN AND THE RESULTS ACHIEVED

The 10 CFR 50.59 process procedure was revised effective April 1, 1997. Rigorous training of evaluators and reviewers, which is still in progress, has already resulted in comprehensive and detailed 10 CFR 50.59 evaluations. In addition, the guidance for conducting SA/USQDs has been centralized with all organizational units using the same procedure for guidance, training and format.

Training/Qualification requirements have been established for personnel performing and reviewing 10 CFR 50.59 evaluations. Personnel training began in March 1997 and will continue until sufficient personnel are certified. Only personnel who have been trained and qualified under the new program will be authorized to perform or review 10 CFR 50.59 evaluations after June 30, 1997. In the interim, the Safety Analysis Group is reviewing 10 CFR 50.59 evaluations to ensure high quality.

In addition, FPC has taken the following specific corrective action to address Violation A.3.:

Administrative instruction AI-400C (New Procedures and Procedure Change Process) dated 03/31/97, provides specific requirements for ensuring proper reviews of operating procedures during the revision cycle, including the requirement for reviews by system engineering. The system engineer ensures that design engineering reviews the procedure if the change may affect the design basis or engineering basis of the plant.

These corrective actions provide reasonable assurance that similar violations will not recur.

CORRECTIVE ACTIONS THAT WILL BE TAKEN TO AVOID FURTHER VIOLATIONS

FPC will amend the Improved Technical Specification Basis and update the FSAR to include the swing "B" makeup pump (MUP-1B) as being a selected Engineering Safeguards load. The update will be completed by November 20, 1997.

VIOLATION A.4.

CORRECTIVE ACTIONS THAT HAVE BEEN TAKEN AND THE RESULTS ACHIEVED

The 10 CFR 50.59 process procedure was revised effective April 1, 1997. Rigorous training of evaluators and reviewers, which is still in progress, has already resulted in comprehensive and detailed 10 CFR 50.59 evaluations. In addition, the guidance for conducting SA/USQDs has been centralized with all organizational units using the same procedure for guidance, training and format.

Training/Qualification requirements have been established for personnel performing and reviewing 10 CFR 50.59 evaluations. Personnel training began in March 1997 and will continue until sufficient personnel are certified. Only personnel who have been trained and qualified under the new program will be authorized to perform or review 10 CFR 50.59 evaluations after June 30, 1997. In the interim, the Safety Analysis Group is reviewing 10 CFR 50.59 evaluations to ensure high quality.

Interdisciplinary and interdepartmental reviews have been established through use of Design Review Boards (DRBs). Depending on the complexity of the modification, formal, structured and extensive conceptual, final and other design reviews are regularly performed. Third party independent design reviews have been and will continue to be obtained for a number of significant design changes to assist the DRBs.

Modifications are currently required to have a complete 10 CFR 50.59 evaluation even if the established screening requirements would result in none required. Under limited conditions, the Design Engineering Manager may waive the requirement for this 10 CFR 50.59 safety evaluation if it is clear none is required.

CORRECTIVE ACTIONS THAT WILL BE TAKEN TO AVOID FURTHER VIOLATIONS

FPC is currently installing an emergency feedwater flow limiting venturi modification. The completion is scheduled for November 28, 1997.

VIOLATION A.5.

CORRECTIVE ACTIONS THAT HAVE BEEN TAKEN AND THE RESULTS ACHIEVED

The 10 CFR 50.59 process procedure was revised effective April 1, 1997. Rigorous training of evaluators and reviewers, which is still in progress, has already resulted in comprehensive and detailed 10 CFR 50.59 evaluations. In addition, the guidance for conducting SA/USQDs has been centralized with all organizational units using the same procedure for guidance, training, and format.

Training/Qualification requirements have been established for personnel performing and reviewing 10 CFR 50.59 evaluations. Personnel training began in March 1997 and will continue until sufficient personnel are certified. Only personnel who have been trained and qualified under the new program will be authorized to perform or review 10 CFR 50.59 evaluations after June 30, 1997. In the interim, the Safety Analysis Group is reviewing 10 CFR 50.59 evaluations to ensure high quality.

Interdisciplinary and interdepartmental reviews have been established through use of Design Review Boards (DRBs). Depending on the complexity of the modification, formal, structured and extensive conceptual, final and other design reviews are regularly performed. Third party independent design reviews have been and will continue to be obtained for a number of significant design changes to assist the DRBs.

Modifications are currently required to have a complete 10 CFR 50.59 evaluation even if the established screening requirements would result in none required. Under limited conditions, the Design Engineering Manager may waive the requirement for this 10 CFR 50.59 safety evaluation if it is clear none is required.

CORRECTIVE ACTIONS THAT WILL BE TAKEN TO AVOID FURTHER VIOLATIONS

FPC will modify the automatic steam supply valve (ASV-204) to the emergency feed pump, EFP-2. The completion of the modification is scheduled for November 28, 1997.

VIOLATION A.6.

CORRECTIVE ACTIONS THAT HAVE BEEN TAKEN AND THE RESULTS ACHIEVED

The FSAR was revised in Revision 23, dated November 18, 1996, Section 4.3.10.1, Boron Dilution, to change information on the boron precipitation methods following a LOCA, reversing the preferred and backup methods to be consistent with the current CR-3 licensing basis.

CORRECTIVE ACTIONS THAT WILL BE TAKEN TO AVOID FURTHER VIOLATIONS

Boron precipitation continues to be a priority issue for the Babcock and Wilcox Owners Group (B&WOG). Framatome Technologies, Inc. (FTI, previously Babcock and Wilcox) has communicated with appropriate NRC staff on this issue and has documented the technical basis for the positions communicated to the NRC in December 1992. The NRC responded to these positions with its March 9, 1993 letter (A. Thadani, NRC, to P. S. Walsh, B&WOG). In July 1996, the B&WOG presented additional analyses regarding boron precipitation under SBLOCA conditions to the NRC. The B&WOG made a formal submittal of the analyses to the NRC in a letter dated March 27, 1997. The B&WOG is moving forward on long-term plans to resolve this issue.

FPC's license contains a condition which requires the decay heat drop line flow indication to be operable in order to be in full compliance. The need for drop line flow indication has been evaluated and found to be unnecessary. FPC is developing a license change request to resolve this issue with NRC approval anticipated prior to restart.

DATE WHEN FULL COMPLIANCE WILL BE ACHIEVED

a. Full compliance with the programmatic issues of these violations was achieved with the implementation of the new 10 CFR 50.59 process.

b. Full compliance with the above EFW system and EDG loading issues will be achieved by physical modification of the plant and amendment of the license. The modifications and amendments will be completed prior to restart.

c. The procedural changes will be accomplished as part of Restart Issues OP-2 through -6, which will be completed prior to restart.

d. Full compliance with the boron precipitation requirements of the license will be achieved with the approval of the license amendment prior to restart.

VIOLATION B.

10 CFR 50, Appendix B, Criterion III, Design Control, requires, in part, that measures be established to assure that applicable regulatory requirements and the design basis, as defined in 10 CFR 50.2, Definitions, and as specified in the license application, are correctly translated into specifications, procedures, and instructions. In addition, 10 CFR 50, Appendix B, Criterion III, requires that design control measures provide for verifying or checking the adequacy of design by individuals other than those who performed the original design. It also requires that design control measures shall be applied to items such as the following: reactor physics, stress, thermal, hydraulic, and accident analyses. The licensee's Quality Program commitments, as described in Table 1-3 of the FSAR, states that in all cases, the design verification shall be completed prior to relying on the component, system, or structure to perform its safety-related function.

Contrary to the above, measures were not established to assure that applicable regulatory requirements and the design basis were correctly translated into specifications, procedures and instructions in the following examples:

1. The design basis information from calculation E91-0026, approved by the licensee's engineering group on May 11, 1989, was not adequately translated into design documents, in that, the FSAR, Enhanced Design Basis Document, and the ITS BASES were not updated to state that the turbine-driven EFW pump (EFP-2) was assumed to be running when the motor-driven EFW pump (EFP-1) tripped automatically at 500 psig reactor coolant system pressure.

2. Design basis information was not correctly translated into the design input requirements for modification package 96-04-12-01, "ASV-204 EFIC Auto Open Removal," in that the previous credit being taken for EFP-2 operating after EFP-1 automatically tripped when RCS pressure decreased to 500 psig during a LOCA concurrent with a loss of offsite power (LOOP) and failure of the train B vital battery was not recognized in the preparation of the modification package. As a result, modification package 96-04-12-01, which was installed in May 1996, removed the train "A" EFW initiation and control (EFIC) automatic open signal from valve ASV-204, one of the two steam supplies to EFP-2, which would have prevented EFP-2 from automatically starting during certain accident scenarios. The design basis was not met from May 1996 through September 1996. In an event, there would have been no EFW for the period of time between the 500 psig actuation signal and when RCS pressure is reduced below the low pressure injection pump shutoff head (approximately 185 psig), when EFW is no longer required for residual heat removal.

3. On December 6, 1994, the design basis was not correctly translated into procedures in that, Calculation M94-0056, performed to generate Procedure OP-103B, Curve 15, Nuclear Closed Cycle Cooling System (SW) Heat Exchanger Fouling versus Ultimate Heat Sink (UHS) Temperature, did not correctly model the heat input to the SW Heat Exchangers from the Reactor Building Fan Coolers. As a result, Curve 15 allowed a larger number of SW Heat Exchanger tubes to be blocked, which could have resulted in the SW Heat Exchanger outlet temperature exceeding the 110° F limit during accident conditions and the system not being capable of removing design basis heat from safety-related equipment.

4. Regulatory requirements were not translated into procedures and the licensee failed to provide measures to verify the adequacy of design by an individual other than those who performed the original design. Specifically, Engineering Procedure NEP-210, Modification Approval Records, Rev. 15, dated January 16, 1996, was inadequate in that it allowed unverified calculations to be relied upon to support modification installation and return to service. As a result, REA 96-047, EDG Loading Case Study, was not verified and was used to support modification package 96-04-12-01 approval in April 1996 which contributed to the introduction of three USQs related to EDG loading.

5. As of June 5, 1996, design basis information was not correctly translated into ITS Surveillance Procedures (SP) SP-324, Containment Inspection, SP-341, Monthly Containment Isolation Valve Operability Check, and SP-346, Containment Penetrations Weekly Check During Refueling Operations. Engineering Procedure NEP-210, Modification Approval Records was inadequate in that it did not provide sufficient guidance to incorporate containment isolation valve surveillance requirements in the review of modifications and calculations. In 1988, 1990, and 1996, modifications were

installed that would have required revisions to SP-324, 346, and 341, to include certain valves/blind flanges. In 1991, a reanalysis of two containment penetrations resulted in reclassification of the penetrations such that a revision to SP-341, to include valves/blind flanges in the procedure was necessary. In 1996, a review of surveillance compliance to ITS requirements regarding containment integrity was conducted. This review failed to consider the surveillance requirements for containment penetrations in the context of maintenance conditions in SP-346. As a result of the modifications, reanalysis and review of surveillance compliance, SP-341 did not include 18 valves/blind flanges in the monthly performance check; SP-324 did not include nine valves/blind flanges in the mode 4 to mode 5 surveillance requirement; and SP-346 did not include 55 valves/blind flanges in the surveillance requirement.

This is a Severity Level III violation (supplement I).

ADMISSION OR DENIAL OF THE ALLEGED VIOLATION

FPC accepts the violation.

REASON FOR VIOLATION

This violation resulted from weaknesses in implementing engineering program requirements. Lack of accountability by some engineering personnel resulted in inadequate adherence to established engineering policies.

CORRECTIVE ACTIONS THAT HAVE BEEN TAKEN AND THE RESULTS ACHIEVED

As discussed at the enforcement conference and FPC/NRC public meetings, FPC's MCAP II program includes comprehensive corrective actions to address the areas of concern in engineering programs. FPC has implemented a number of corrective actions associated with both the ongoing MCAP II activities and other restart issues discovered since the issuance of MCAP II. The following is a summary of the comprehensive corrective actions taken to strengthen the engineering programs:

Organizational

To increase the organizational effectiveness and management oversight, the engineering organization has had a number of structural changes, manager and supervisor reassignments, and increased staff levels. Of key interest is a new management position for engineering programs (ISI/IST, new/finite duration efforts, Appendix R, etc.) and the transfer of the safety analysis group back to Nuclear Operations Engineering to focus on safety analysis/10 CFR 50.59 activities. To supplement the traditional design engineering background, several positions have been filled with individuals that have strong experience in other nuclear fields (Operations, Licensing, Safety Analysis).

Teamwork/Interfaces

Increasing the level of teamwork both within engineering and with other departments has been addressed through several actions. These include requirements for operations, systems engineering, and design personnel to be jointly involved during the calculation development/revision process, increasing the use of project teams for significant modifications, and specific guidance for internal interdisciplinary interface within the design engineering groups. In addition, Design Review Boards (DRBs) ensure increased participation by other plant departments during the modification development phase, for certain significant modifications, resulting in improved final design packages.

Expectations/Communications

To improve implementation of the engineering processes, expectations for engineering management and staff were developed and communicated to engineering personnel. These expectations support emphasis on plant safety and set a high standard for engineering performance. The standards cover areas such as 10 CFR 50.59 evaluations, DRBs, and Analysis/Calculation performance. Communications and reinforcement of these expectations and other key informational facts such as plant priorities are occurring through departmental/group meetings, visual displays, and formal documentation.

Process/Procedures

The recent 10 CFR 50.54(f) response stated that the design control program was basically sound and that procedures are in place that address regulatory requirements. However, the changing organization and increased staffing require that a greater level of detail and guidance be provided in the design control procedures. To this end, changes have been made to the design control procedures to address the areas where implementation was not up to FPC's standard. The areas of enhanced procedural guidance include 10 CFR 50.59 evaluations, engineering interdisciplinary interface, design input considerations, external interfacing for modifications and calculations, analysis/calculation review and verification, timeliness of engineering document (design basis documents, computer data bases) updates, design verification process, and engineering review prior to return to service of modifications.

FPC is confident that these organizational and management/staff changes, improvement in teamwork, clear performance expectations, and an increased level of detail and procedural guidance for the design control program have already substantially improved engineering performance.

Specific corrective actions taken for the violation include the following:

1. FPC guidance has been enhanced with a number of changes to our design expectations, controls and processes that, taken in the aggregate, make it much less likely that such an error would occur.

2. Plant and procedure modifications have been implemented to resolve the concerns, with EDG load management and EFW requirements for small break loss of coolant accident (SBLOCA) mitigation, that led to modification package 96-04-12-01. Diverse initiation from either DC train will be restored by completing the ASV-204 modification.

3. Procedure OP-103B, Curve 15, Nuclear Closed Cycle Cooling System (SW) Heat Exchanger Fouling versus Ultimate Heat Sink (UHS) Temperature, and its supporting calculation (M94-0056) have been revised.

4. Interim guidance for verification of case studies and Request for Engineering Assistance (REA) responses, which are used as design inputs for plant modifications, was issued on October 11, 1996 and formal procedure changes were made on March 31, 1997. Restart Issue OP-6 was developed to address NEP guidance and design controls. It contains a number of actions that have been or will be taken to address these issues.

5. Containment isolation valves (CIVs) and blind flanges within the containment penetration barriers necessary for containment isolation were validated for Modes 1 through 4 operation to ensure proper configuration controls are in place.

6. NEP-210, Modification Approval Records, and NEP-254, Plant Equipment Equivalency Replacement Evaluation, have been revised to provide guidance to design engineers for items affecting containment integrity.

CORRECTIVE ACTIONS THAT WILL BE TAKEN TO AVOID FURTHER VIOLATIONS

FPC is confident that the improved programs implementing our 10 CFR 50.59 process, our design control program, and our corrective action program including the associated training will ensure self-identification and evaluation of design deficiencies or potential USQs. In addition, prior to plant restart, FPC is engaged in an extensive System Readiness Review Program to assure that safety-related systems are in compliance with the licensing and design basis of the CR-3 facility.

Specific corrective actions that will be taken for the violation include the following:

1. To resolve examples 1 and 2, the FSAR, EDBD, and the ITS BASES will be corrected prior to startup. Expectations regarding the use of the FSAR, EDBD, and the ITS BASES to provide independent input and oversight have been significantly strengthened.

2. No further corrective actions are required to resolve example 3.

3. To resolve example 4;

   a. FPC will address Nuclear Engineering Procedure guidance and design control issues by September 15, 1997.

   b. Plant and procedure modifications have been implemented to resolve the concerns, with EDG load management and EFW requirements for small break loss of coolant accident (SBLOCA) mitigation, that led to modification package 96-04-12-01. Diverse initiation from either EFIC train will be restored by completing the ASV-204 modification.

c. The generic issue of the extent of condition regarding plant design is being addressed by performing system readiness reviews. The system readiness reviews will be complete by October 27, 1997. These reviews are intended to provide reasonable assurance of conformity of the system and its operation to the design basis and licensing basis.

d. The diesel loading calculation will be completed by November 28, 1997.

4. The following corrective actions will be taken to resolve example 5.

   a. SP-324, Containment Inspection (for ITS 3.6.3.4), and SP-341 (for ITS 3.6.3.3), Monthly Containment Isolation Valve Operability Check, will be revised prior to restart to include additional CIVs as part of a modification package being completed prior to restart.

   b. SP-346 (for ITS 3.9.3.1), Containment Penetrations Weekly Check During Refueling Operations, will be revised to include appropriate CIVs and to address leakage pathways potentially created during such outages that are not addressed in Table 5-9.

   c. A new series of drawings to clearly identify the penetration configurations is being developed. These drawings will be completed after the modification, discussed in a. above, has been completed and will be issued by August 11, 1997.

DATE WHEN FULL COMPLIANCE WILL BE ACHIEVED

The following discussions correspond to the individual violation examples:

1. For examples 1 and 2, full compliance will be achieved when the FSAR, EDBD, and the ITS BASES are revised to reflect the design and licensing basis, prior to restart.

2. Full compliance for example 3 was achieved when Procedure OP-103B, Curve 15, and its supporting calculation were revised.

3. Full compliance for example 4 will be achieved when Restart Issue OP-6 corrective actions are completed prior to restart.

4. For example 5, compliance was achieved for NEP-210 and NEP-254 with revisions issued on March 31, 1997. Additionally, surveillance procedures will be appropriately revised prior to their required usage. Full compliance for example 5 will be achieved upon revision of SP-346 prior to Refuel 11 refueling operations.

VIOLATION C.

10 CFR 50, Appendix B, Criterion XVI, Corrective Action, requires that measures be established to assure that conditions adverse to quality, such as non-conformances, are promptly identified and corrected. In the case of significant conditions adverse to quality, the

measures shall assure that the cause of the condition is determined and the corrective action taken to preclude repetition.

Contrary to the above, the licensee failed to correct conditions adverse to quality and failed to take measures to assure that corrective actions were taken to preclude repetition of significant conditions adverse to quality as follows:

1. Precursor Card 96-2750, dated May 31, 1996, and Problem Report 96-0210, dated July 3, 1996, identified that changes made to the facility in April 1996 introduced EDG loads that were in excess of the ITS limits, a significant condition adverse to quality; however, adequate corrective actions were not implemented. The adverse conditions were not corrected as of October 11, 1996. As a result, the plant operated for several months with USQs related to EDG loading.

2. Problem Report 94-0218, dated June 24, 1994, described a problem where engineers failed to address EDG loading effects of several modifications in the 10 CFR 50.59 evaluations; however, the licensee failed to take adequate corrective actions for this significant condition adverse to quality. As a result, in April 1996, the 10 CFR 50.59 evaluation for modification package 96-04-12-01 did not address EDG loading effects and modification package 96-14-12-01 was inappropriately installed and placed in operation with USQs.

3. On October 12, 1994, the licensee identified that penetrations were not being tested in accordance with ITS 3.6.3.3, as reported in Licensee Event Report 94-007; however, the corrective actions taken for LER 94-007, dated November 10, 1994, were not adequate to prevent recurrence, resulting in numerous additional valves/blind flanges that were omitted from the surveillance procedures being identified in 1996.

ADMISSION OR DENIAL OF THE ALLEGED VIOLATION

FPC accepts the violation.

REASON FOR VIOLATION

As discussed at the pre-decisional enforcement conference, FPC's corrective action program had not consistently ensured timely follow-up to identified problems. Therefore, the reasons for the violation were an inadequate corrective action process and lack of appropriate line management oversight and expectations for the program.

CORRECTIVE ACTIONS THAT HAVE BEEN TAKEN AND THE RESULTS ACHIEVED

FPC has implemented a revised corrective action program which includes a single graded approach process (priority classification), a screening committee, root cause teams, and a Corrective Action Review Board (CARB). A "Line Management Owned,"single-graded approach corrective action process is now in place. The revised process ensures that Precursor Cards are promptly reviewed, prioritized, and distributed to the assigned owner. This has resulted in improvements of timely cause determinations and corrective action implementation.

Clear lines of ownership, responsibility, and accountability have been established in the new corrective action process. The scope of investigations has been broadened to include the determination of the Extent of Condition. Among the continuing enhancements to the corrective action process, the procedure requires that the investigation address the extent of condition, and that the responsible line manager conduct an effectiveness review of the associated corrective actions to ensure prevention of recurrence.

A series of work stand-downs and accountability sessions have been held to ensure management expectations regarding a questioning attitude are understood. Results have shown an increased sensitivity by the engineering staff demonstrated by an increasing number of precursor cards submitted by engineering personnel. The issues being raised show that the threshold of issue identification has lowered and that the engineers are demanding answers and results.

The CARB is responsible for validating condition significance and approving extensions for significant precursor investigations and corrective action due dates. The CARB is specifically charged with ensuring that corrective actions are supported and directly related to the identified root causes and will reduce recurrence rate in a timely manner. The CARB is intended to provide an additional level of management oversight. Identification, cause determinations, and associated corrective actions have become more comprehensive since this process was initiated in November of 1996.

FPC has also implemented a revised 10 CFR 50.59 process. The response to Violation A details the process improvement actions taken and the results achieved.

<u>CORRECTIVE ACTIONS THAT WILL BE TAKEN TO AVOID FURTHER VIOLATIONS</u>

The following corrective actions will be taken to avoid further violations:

1. The Quality Programs organization will continue to monitor the new process to ensure proper implementation. Additional personnel are being trained on the new 10 CFR 50.59 process to expand the base of knowledgeable personnel onsite.

2. Additional programmatic actions addressing engineering performance issues are in MCAP II and the Restart Plan. Further corrective action process enhancements and training on the new 10 CFR 50.59 process will be implemented as experience and effectiveness assessments warrant.

<u>DATE WHEN FULL COMPLIANCE WILL BE ACHIEVED</u>

Full compliance has been achieved with the implementation of the new corrective action process and the new 10 CFR 50.59 process.

<u>ADDITIONAL INFORMATION REGARDING THESE VIOLATIONS</u>

The issues identified by these violations are being tracked by the CR-3 corrective action program. Where required, Licensee Event Reports (LERs) have been submitted which discuss specific reportable events related to these issues.

Attachment 2

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

RESPONSE SECTION	COMMITMENT	DUE DATE
Pg. 4	The plant will not be returned to service until the issues identified in the October 28, 1996 letter are satisfactorily resolved.	Prior to Restart
Pg. 5	Only personnel who have been trained and qualified under the new program will be authorized to perform or review 10 CFR 50.59 evaluations after June 30, 1997.	June 30, 1997
Pg. 5	In the interim, the Safety Analysis Group is reviewing 10 CFR 50.59 evaluations to ensure high quality.	In Progress
Pg. 5	Modifications are currently required to have a complete 10 CFR 50.59 evaluation even if the established screening requirements would result in none required. Under limited conditions, the Design Engineering Manager may waive the requirement for this 10 CFR 50.59 safety evaluation if it is clear none is required. This process will continue until formal training and establishment of qualified reviewers for 10 CFR 50.59 evaluation is complete.	In Progress
Pg. 5	Third party independent design reviews have been and will continue to be obtained for a number of significant design changes.	In Progress
Pg. 5	FPC will provide training on the application, interpretation, and expectations of Improved Technical Specifications (ITS) and ITS BASES in support of the design process to the Nuclear Operations Engineering Staff as a continuing effort.	December 31, 1997

Pg. 6	FPC has committed to resolve these issues, which will be complete prior to restart, including licensing submittals and receipt of NRC approvals for amendments to the Crystal River Unit 3 (CR-3) License and ITS amendments. The program will include modifications to resolve the individual USQs identified in the violations and restore adequate design margins. These include modifications such as uprating the EDG, installing cavitating venturis in the EFW system, making EFIC improvements, and completing the ASV-204 modification.	Prior to Restart
Pg. 6	Prior to plant restart, FPC is engaged in an extensive System Readiness Review Program to assure that safety-related systems are in compliance with the licensing and design basis of the CR-3 facility.	October 27, 1997
Pg. 7	Procedural changes will be made to improve the definition of the responsibilities of the design engineer, verification engineer, and the EDG Load Management Program, and the verification of calculations. This enhanced procedural guidance will include 10 CFR 50.59 evaluations, engineering interdisciplinary interface, design input considerations, external interfacing for modifications and calculations, analysis/calculation review and verification, timely updating of engineering documents (design basis documents, computer data bases), design verification process, and engineering review prior to return to service of modifications. The procedural changes will be accomplished as part of Restart Issues OP-2 through -6, which will be completed prior to restart.	September 15, 1997
Pg. 8	The Final Safety Analysis Report (FSAR), Enhanced Design Basis Document (EDBD), and Analysis Basis Document (ABD), will be updated to show EDG and EFW equipment dependencies and equipment limitations.	November 20, 1997

Pg. 8	Design basis training, including design assumptions on equipment availability and limitations, will be provided to targeted Operations personnel.	December 31, 1997.
Pg. 8	The EDGs are being uprated to establish design margin over the current loading.	Prior to Restart
Pg. 8	License amendments will be submitted to revise the design and licensing basis and the ITS BASES to reflect this uprate prior to restart from the present shutdown.	Prior to Restart
Pg. 8	The emergency operating procedures (EOPs) are being revised as part of the EOP Enhancement Program. The revised procedures will have appropriate reviews by design personnel and will be issued prior to restart from the present outage.	Prior to Restart
Pg. 9	FPC will amend the Improved Technical Specification Basis and update the FSAR to include MUP-1B as being a selected Engineering Safeguards load.	November 20, 1997
Pg. 9	FPC is currently installing an emergency feedwater flow limiting venturi modification. The completion is scheduled for November 28, 1997.	November 28, 1997
Pg. 10	FPC will modify the automatic steam supply valve (ASV-204) to the emergency feed pump, EFP-2. The completion of the modification is scheduled for November 28, 1997.	November 28, 1997

Pg. 11	FPC's license contains a condition which requires the decay heat drop line flow indication to be operable in order to be in full compliance. The need for drop line flow indication has been evaluated and found to be unnecessary. FPC is developing a license change request to resolve this issue with NRC approval anticipated prior to restart.	Prior to Restart
Pg. 15	To resolve examples 1 and 2, of Violation B, the FSAR, EDBD, and the ITS BASES will be corrected prior to startup.	Prior to Restart
Pg. 16	The diesel loading calculation will be completed by November 28, 1997.	November 28, 1997
Pg. 16	SP-324, Containment Inspection (for ITS 3.6.3.4), and SP-341 (for ITS 3.6.3.3), Monthly Containment Isolation Valve Operability Check, will be revised prior to restart to include additional CIVs.	August 11, 1997
Pg. 16	SP-346 (for ITS 3.9.3.1), Containment Penetrations Weekly Check During Refueling Operations, will be revised to include appropriate CIVs and to address leakage pathways potentially created during such outages that are not addressed in Table 5-9.	Prior to next refueling
Pg. 16	A new series of drawings to clearly identify the penetration configurations is being developed. These drawings will be completed after the modification, discussed in a. above, has been completed and issued by August 11, 1997.	August 11, 1997
Pg. 18	The Quality Programs organization will continue to monitor the new process to ensure proper implementation.	In Progress
Pg. 18	Further corrective action process enhancements and training on the new 10 CFR 50.59 process will be implemented as experience and effectiveness assessments warrant.	In Progress