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

October 28, 1996
3F1096-22

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

Subject: Crystal River Unit 3 Forced Outage

Dear Sir:

On September 2, 1996, Florida Power Corporation (FPC) shut down the Crystal River Unit 3 (CR-3) nuclear plant due to a leak in the turbine lube oil system. During this forced outage, FPC determined that a modification had been made to the plant during the Spring, 1996 Refuel 10 outage which created an Unreviewed Safety Question (USQ) regarding emergency diesel generator (EDG) loading. This USQ involved a reduction in the margin of safety described in portions of the Technical Specification Bases.

On October 4, 1996, while still shut down, FPC was preparing a submittal to request NRC approval of a license amendment to change the affected EDG Technical Specification Bases when additional questions arose regarding the change to the emergency feedwater (EFW) system which created the diesel loading USQ. These questions involved failure modes with the EFW system which needed to be evaluated to ensure the system could perform its safety function and reliance on the turbine-driven, "B" train emergency feedwater pump for "A" train EDG load management. Due to the EFW/EDG issues, and some other design-related issues, FPC management made a decision to keep CR-3 shut down until these issues are adequately addressed. The purpose of this letter is to inform the NRC of our plans to address these issues prior to restarting the plant.

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The issues described in the attached list were identified through a review conducted by a multi-discipline team involved in reviewing the Emergency Operating Procedures (EOPs) and through design reviews by the engineering organization. The list was reviewed by CR-3 senior management and the items are considered necessary to ensure safety system operability or to increase design margins. Each issue has been documented in the CR-3 corrective action system and will be tracked to closure. Several of the issues have been determined to be reportable and Licensee Event Reports are being processed.

FPC will ensure the safety systems in question are capable of performing their design basis functions prior to restart from this outage. As an added level of assurance, FPC will be establishing an internal restart panel which will function similar to an NRC restart panel using NRC Inspection Manual 0350 as a guideline for conducting the restart readiness review. Upon completion of the work to resolve the issues, the panel will conduct a final review to confirm that all issues have been resolved adequately. When satisfied, restart of the unit will be recommended to the Senior Vice President, Nuclear Operations. In addition, the Nuclear General Review Committee (NGRC) will conduct an independent review prior to restart.

Project teams or individual lead responsibility have been established for each issue to support the design, licensing and installation activities necessary to complete the outage work scope. Final resolutions for some of the issues on the list have not yet been determined. Other resolutions require relatively long lead procurement activities. Therefore, an integrated outage schedule is not available at this time. However, we expect the unit to remain shutdown until at least mid-January, 1997. This will also likely move our next refueling outage, Refuel 11, to the fall of 1998 rather than the spring of 1998, as currently scheduled. The NRC will be kept abreast of the schedule and progress on these issues as the outage continues.

Sincerely,

P. M. Beard, Jr.
Senior Vice President
Nuclear Operations

PMB/BG

Attachment

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

CR-3 Design Margin Improvement Outage Scope of Work

1. High Pressure Injection (HPI) Pump Recirculation to the Makeup Tank

Concern: The HPI pumps draw suction from the Borated Water Storage Tank (BWST) during the initial phase of emergency core cooling system (ECCS) injection. Once BWST level has reached a pre-determined level, suction is switched to the reactor building sump with the HPI pumps taking suction from the discharge of the low pressure injection (LPI) pumps (piggyback operation). During piggyback operation, LPI pump discharge pressure keeps the check valve in the suction line from the makeup tank (MUT) to the HPI pumps closed (MUV-65). During long term small break LOCA (SBLOCA) cooling, HPI flow may require throttling due to lower required ECCS flow. If throttling continues, procedures will eventually direct the operators to increase total HPI pump flow by opening the HPI recirculation valves at a pre-determined flow rate to divert some flow to the MUT. Since no flow is exiting the MUT, the tank could fill up with recirculation flow and lift the relief valves, dumping fluid onto the auxiliary building floor. This would result in the transfer of RB sump fluid to the auxiliary building sump, which reduces the amount of water available in the RB sump from which the LPI and reactor building spray pumps take suction during the later stages of core and containment cooling. This could also create a release path for post accident radioactive fluid outside containment.

Resolution: FPC is consulting with Framatome Technologies, Inc. (FTI) to confirm whether the scenario is valid and within the CR-3 design basis. Although the resolution of this issue is still undetermined at this time, preliminary indications are that opening of a high point vent valve may preclude the need to open the HPI recirculation valves in the SBLOCA scenarios of concern.

Schedule: This issue will be resolved prior to startup from the current outage.

2. HPI System Modifications to Improve SBLOCA Margins

Concern: The CR-3 HPI system currently meets all design and licensing basis functional requirements. However, the CR-3 configuration is not consistent with the designs at other Babcock and Wilcox (B&W) plants. As a result, HPI minimum and maximum flow limits are more restrictive and peak cladding temperatures for certain SBLOCA scenarios are higher. In addition, the reduced system design margin has created the need for several manual operator actions to ensure adequate core cooling. FPC intends to reduce the operator burden created by these actions and the system margin deficit through hardware modifications. These modifications would also make the CR-3 HPI system design more like other B&W plants.

Resolution: At this time, the following modifications are being considered:

a. Installing cavitating venturis to limit flow through any single injection leg due to a postulated break in that leg.

b. Installing cross-tie piping downstream of the HPI injection control valves to deliver increased core cooling flow should a failure prevent one or more of the injection valves from opening.

c. Modifying the normal makeup line to ensure automatic isolation occurs upon ES actuation to eliminate the operator action now required to perform this function. This involves modifying the power supply to the existing isolation valve (MUV-27) and adding another isolation valve powered from the opposite train in series with MUV-27. (Note: the proposed installation of the cavitating venturis could preclude the need for this modification).

Schedule: Since the HPI system is fully capable of meeting its design function, these modifications are not considered necessary to complete during the current outage. However, FPC is developing the design packages and determining whether equipment can be procured in a time frame to install in the current outage given the schedules for other activities.

3. <u>LPI Pump Mission Time</u>

Concern: During the IPAP inspection, an issue was raised regarding the need to establish flow through the decay heat removal (DH) drop line to the decay heat removal (LPI) pumps as part of small break LOCA mitigation. CR-3 has two redundant, independent LPI trains which can take suction from the RB sump during long term recirculation core cooling. However, certain small break LOCAs could result in long-lasting, elevated RCS pressures such that the LPI pumps would have to operate in the piggyback mode at low flow rates for an extended period of time. As that period of time approaches the current low flow mission time for the LPI pumps, plant procedures direct the operators to trip one pump and open the DH drop line valves to the RB sump to provide additional flow through the remaining running LPI pump. There is only one DH drop line at CR-3 (and many other pressurized water reactors) which has three motor-operated valves in series. Failure of any one of the drop line valves to open would prevent flow through the line. If the DH drop line was necessary to fulfill the ECCS long term core cooling function for small break LOCA mitigation, this would violate the single failure design criterion.

Resolution: The concern described above is time-dependent. If the time frame is long enough after the event, opening of the DH drop line could be considered a long-term recovery action as opposed to an emergency core cooling function. FPC considers the long term recovery phase beyond the time frame implied by the regulations where applying the single failure design criterion is necessary. At the time of the

IPAP inspection, the low flow mission time for the LPI pumps was 72 hours, which was questionable from an ECCS versus long term recovery perspective. FPC is currently low-flow testing a pump which is identical to the CR-3 LPI pumps. The test flow rate is approximately 100 gallons per minute (gpm). The design flow rate of the LPI pumps is 3000 gpm. The results of this test are expected to prove that the pumps could run for an extended period at very low flows without damage. If the test is successful, procedures will be revised to characterize opening the DH drop line in this scenario as a long term recovery action rather than an ECCS function.

Schedule: This issue will be resolved before startup from the current outage. As of 3:30 p.m. on October 25, 1996, the pump had completed 18 days of continuous low-flow testing with no performance (head curve) degradation, no mechanical seal leakage, no indication of unexpected bearing wear, and all vibration parameters stable and well below the action levels specified in the surveillance procedure. The testing is continuing beyond 18 days.

4. Reactor Building Spray Pump 1B NPSH

Concern: During the long term recirculation phase of core and containment cooling, the reactor building spray pumps (BSPs) take suction from the reactor building sump. Calculations have shown BSP-1B to have little margin between required and available net positive suction head (NPSH) during this phase of operation. A recent revision of the calculation shows the margin to be approximately one foot of water. It is desired to increase this margin.

Resolution: FPC currently plans to conduct factory testing and/or modify the pump impeller to improve the margin between required and available NPSH.

Schedule: This issue will be resolved before startup from the current outage.

5. Emergency Feedwater System Upgrades and Diesel Generator Load Impact

Concern 5.1: The CR-3 EFW system is comprised of two 100% capacity trains, with the "A" train pump (EFP-1) being motor driven and the "B" train pump (EFP-2) being steam driven. The steam for the EFP-2 turbine driver is fed through redundant inlet valves (ASV-5 and ASV-204) to ensure the availability of steam given a failure of one of the inlet valves to open. Each pump feeds both steam generators. For a portion of the flow path from the emergency feedwater tank (EFT-2), the two pumps share a common suction line. Under certain accident scenarios, there are failure modes which can cause the calculated NPSH available to both pumps to be less than required. For example, a failure of the DC control power source for the injection control valves in one train of EFW can result in the pump in that train producing high flows which result in excessive friction head losses through the common suction line.

Concern 5.2: Motor-driven EFP-1 is powered from the "A" train ES bus and is connected to the "A" emergency diesel generator (EGDG-1A). EFP-2 is steam driven and therefore does not affect "B" train EDG loading. However, portions of the load management scheme for EGDG-1A depend on the availability of EFP-2 to: 1) limit the total flow produced by EFP-1 during the early stages of diesel loading and 2) permit EFP-1 to be shut down and the "A" train LPI pump (and other engineered safeguards features) to be started in the later stages of accident mitigation. Therefore, some postulated failure modes which cause EFP-2 to be unavailable invalidate assumptions made in EGDG-1A loading calculations and some accident analyses which may have taken credit for flow from EFP-2 after EFP-1 was shut down.

Resolution: At this time, the following modifications are being considered:

a. Installing cavitating venturis in the EFW pump discharge lines to limit flow during the postulated failures which result in the loss of flow control for an EFW train. This will eliminate the NPSH concern.

b. Re-enabling "A" train Emergency Feedwater Initiation and Control (EFIC) system actuation of EFP-2 via automatically opening steam turbine inlet valve ASV-204. This feature was disabled by a modification in Refuel 10 and will be restored to ensure EFP-2 auto-starts given a failure of the "B" side initiate logic or ASV-5.

c. Installing motor operators on cross-tie valves EFV-12 and EFV-13 to allow remote manual opening of these valves. Opening these valves establishes a flow path allowing the pump from one train to feed the steam generators through the injection lines of the other train. This is desirable to ensure the operators can maintain EFW flow control and indication in certain single failure scenarios without requiring local manual valve operation.

Schedule: This issue will be resolved before startup from the current outage. We expect this issue to require additional interaction with the NRC prior to restart.

6. Emergency Diesel Generator Loading

Concern: The rated capacity of EGDG-1A is challenged by the continuous, automatically connected loads as well as the loads that are manually connected in the later stages of accident mitigation. Three concerns were created by the Refuel 10 modification which removed the "A" train EFIC automatic actuation of ASV-204. Calculated peak transient diesel loads were above the 3500 kW maximum engine rating documented in the FSAR and the ITS basis background for LCO 3.8.1, "AC Sources"; calculated peak diesel load at one minute was above the 3100 kW rating discussed in the basis for Surveillance Requirement 3.8.1.11; and the highest single rejected diesel load

discussed in the basis for Surveillance Requirement 3.8.1.8 increased.

Resolution: A combination of three efforts is being pursued to increase the load capability of EGDG-1A. They include an engine power upgrade to increase one or more of the load ratings; removal and/or reduction of connected loads; and improving the accuracy of the kW meters used to display the generators' output. We expect this issue to also require additional NRC interaction prior to restart.

Schedule: This issue will be resolved before startup from the current outage.

7. <u>Failure Modes and Effects of Loss of DC Power</u>

Concern: A number of CR-3 design and operating vulnerabilities have been identified on a case-by-case basis through design and EOP reviews postulating the effects of a loss of DC power. The loss of DC power also causes a consequential loss of emergency AC power since DC power is required for emergency diesel generator field flashing and bus breaker closure. A Failure Modes and Effects Analysis (FMEA) was performed for the CR-3 Class 1E electrical distribution system (including DC) as part of the original plant design. However, it may not have fully considered system interactions, including effects on redundant trains and components.

Resolution: FPC will perform a DC power FMEA which includes evaluations of system interactions.

Schedule: The FMEA review will be completed to the extent that FPC is satisfied that we have identified any safety significant problems. Such problems will be addressed prior to startup from the current outage.

8. <u>Generic Letter 96-06</u>

Concern: This Generic Letter (GL) identifies three issues regarding the effect of post-accident containment heatup on containment coolers, piping, and penetrations. CR-3 is susceptible to the piping overpressurization phenomenon and is evaluating the water hammer and two-phase heat transfer problems.

Resolution: FPC is installing thermal overpressure protection devices on containment penetrations affected by this phenomenon. Actions to address the impact of the other two issues, if any, will be determined after the review is completed.

Schedule: The overpressure protection devices will be installed prior to startup from the current outage. Actions to address the impact of the other two issues, if any, will be scheduled according to the safety significance of the findings.