December 30, 1976

SUPPLEMENT NO. 3
TO THE
SAFETY EVALUATION REPORT
BY THE
OFFICE OF NUCLEAR REACTOR REGULATION
IN THE MATTER OF
FLORIDA POWER CORPORATION, ET AL
CRYSTAL RIVER UNIT NO. 3
DOCKET NO. 50-302

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1.0 INTRODUCTION

The Nuclear Regulatory Commission's (Commission) Safety Evaluation Report on the matter of the application by Florida Power Corporation, et al to operate the Crystal River Unit No. 3 facility was issued on July 5, 1974. Supplement No. 1 to our Safety Evaluation Report was issued on January 13, 1975, and Supplement No. 2 was issued on December 3, 1976. We concluded in Supplement No. 2 that the Crystal River Unit No. 3 facility may load fuel and be operated in the refueling mode of operation and the cold shutdown mode of operation (Modes 5 and 6 as defined in the plant Technical Specifications), and that upon favorable evaluation of outstanding safety items identified in Supplement No. 2, power operation may be authorized.

On December 3, 1976 the Commission issued an operating license to Florida Power Corporation and eleven co-owners (licensees) authorizing operation of the facility in the refueling mode of operation and the cold shutdown mode of operation.

The purpose of this Supplement to our Safety Evaluation Report is to update the report by providing our evaluation of the outstanding matters identified in Supplement No. 2 and our evaluation of additional information submitted by the licensees since the issuance of Supplement No. 2. This Supplement also discusses our final evaluation of the chemical addition to the containment spray system in Section 6.2.2.

Each of the sections in this Supplement is numbered the same as the section of the Safety Evaluation Report and its Supplements that is being updated, and is supplementary to but not in lieu of the discussion in the Safety Evaluation Report and its Supplements.

The outstanding matters which we stated in Supplement No. 2 that we would address in this Supplement, and the sections in which these matters are discussed, are as follows:

(1) evaluation of the final report regarding the repairs to the containment dome and the structural integrity test of the containment (Section 3.8.1),

(2) evaluation of the engineering hot channel factor for a replacement fuel assembly (Section 4.2.1),

(3) operating procedures and design provisions that will make the likelihood of a pressure transient exceeding the temperature-pressure limits of the reactor pressure vessel acceptably small (Section 5.2),

(4) evaluation of the inservice inspection program for ASME Code Class 1, 2 and 3 components (Section 5.5),

(5) evaluation of the modifications to the containment spray system (Section 6.2.2),

(6) review of the emergency core cooling system performance evaluation (Section 6.3.3),

(7) requirement to modify the pressure sensing lines to the differential pressure transmitters of the reactor coolant system flow indication (Section 7.2), and

(8) evaluation of the provisions for redundant safety grade low water level indication for the borated water storage tank (Section 7.3).

We conclude that all of the matters indicated above have been resolved to the extent that plant operation at power is acceptable within the limitations discussed in Sections 3.8.1 and 6.3.3 of this Supplement. Acceptability of plant operation without these limitations is contingent upon favorable evaluation of (1) the final report of the structural integrity test of the containment dome (Section 3.8.1) and (2) the analysis in regard to the performance of the emergency core cooling system (Section 6.3.3).

3.0 DESIGN CRITERIA - STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.8 Design of Category 1 (Seismic) Structures

3.8.1 Containment

We stated in Supplement No. 2 to our Safety Evaluation Report that we would review the final report related to the structural integrity test of the containment to confirm our conclusion that the repaired containment structure meets the original structural design criteria and will withstand the specified design conditions without impairment of structural integrity or safety function. We also stated that since there has not been any experience with the behavior of such a structure, we required Florida Power Corporation to make a detailed analysis of the repaired dome and to instrument the dome so that a correlation between the predicted and measured behavior could be established when the containment structure is subjected to the structural integrity test.

We have reviewed the information submitted by Florida Power Corporation on December 10, 1976 related to the repair of the containment dome and the structural analysis of the repaired structure. This information, when added to the interim report "Reactor Building Dome Delamination" submitted on June 11, 1976, constitutes the final report on this matter. Based on our review of the final report we conclude that the principal contributor to the delamination of the dome was the lack of radial reinforcement. The concrete alone was not able to support the radial stresses imposed by the tensioning of the tendons.

Florida Power Corporation also submitted its final report, "Reactor Containment Building Structural Integrity Test," GAI Report No. 1930, on December 9, 1976 which presents a description of the test of the containment. In its final report Florida Power Corporation states that the overall response of the structure was well substantiated by the test, and that the displacements observed were within predicted values and were typical of displacements measured on other similar structures with recovery observed to be within normally expected limits for a structure of this type. Florida Power Corporation also concludes that the cracking observed on the dome during the test was slightly greater than would normally be expected in a prestressed dome but substantially less and of smaller magnitude that could be expected in a reinforced dome. Further, the fact that these cracks closed indicated that the structure was still within the elastic range. The strains recorded were also well within the elastic range of the material.

In order to provide assurance that the containment structure will continue to behave as predicted during the life of the plant, we will require Florida Power Corporation to propose modifications to the surveillance program specified in the plant Technical Specifications to include displacement and strain measurements and monitoring of crack patterns and crack widths. We will require that this additional surveillance be in effect at the next schedule surveillance for containment integrity that is specified in Section 4.6.1.6.1 of the plant Technical Specifications. Our principal concern in this regard is the strains that may be introduced as a consequence of temperature differentials across the dome.

Based on our review to date the information provided in the final report of the structural integrity test, we conclude that the plant can be operated within the startup mode 2 at power levels less than five percent of rated thermal power without adversely affecting the health and safety of the public. Our evaluation of our concerns regarding thermal strains and additional surveillance of crack patterns will be discussed in a future supplement to the Safety Evaluation Report.

4.0 REACTOR

4.2 Mechanical Design

4.2.1 Fuel

In Supplement No. 2 to our Safety Evaluation Report we stated that we would report the results of our evaluation of the effects of the higher engineering hot channel factor of the replacement fuel rods compared with the original fuel rods in the replacement fuel assembly identified as fuel assembly 3A33.

We have completed our review of the replacement fuel assembly with regard to the higher engineering hot channel factor. The use of an axially zoned enrichment and density distribution in the replacement fuel rods results in an increase to a hot channel factor of 1.026 for the replacement fuel rods compared with 1.014 for the original fuel rods, which is an increase of 2.6 percent. This increase is compensated for, in part, by the axially zoned loading which tends to flatten the axial power distribution. In addition the replacement fuel assembly will be placed in a location (identified as K-9) of the core where the total peaking factor is approximately 19 percent lower than the maximum peaking factor. Since a margin to the thermal limits exists at the location of the highest peaking factor, the further margin that exists in the location of the replacement fuel assembly therefore provides more compensation than is needed for the higher engineering hot channel factor.

Based on our review and the consideration of the thermal margins as stated above, we conclude that the replacement fuel assembly is acceptable.

5.0 REACTOR COOLANT SYSTEM

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Fracture Toughness

Reactor Coolant System Overpressurization

In Supplement No. 2 to our Safety Evaluation Report we stated that we would evaluate the measures to be taken by Florida Power Corporation to reduce the likelihood of a pressure transient occurring that could result in exceeding the pressure-temperature limits of the reactor pressure vessel.

In a letter to us dated December 2, 1976, Florida Power Corporation has provided an interim response to our request to furnish a description of current administrative procedures and design controls that can be implemented now so as to preclude or minimize the potential for overpressurization.

Crystal River Unit No. 3 utilizes nitrogen gas to maintain a gas bubble in the pressurizer whenever a steam bubble is not maintained, so that no plant operation will involve a solid water condition. Florida Power Corporation will also install a dual setpoint pilot-operated relief valve on the pressurizer. The lower setpoint (550 pounds per square inch) will be initiated by automatic actuation of a temperature switch closing at approximately 300 degrees Fahrenheit temperature of the reactor coolant system during plant cooldown and prior to startup of the decay heat removal system. In the interim until this modification is completed within six months, the control room operator will manually actuate the pilot-operated relief valve by turning a key switch in the non-nuclear instrumentation cabinet upon indication of a pressure of 550 pounds per square inch in the reactor coolant system.

In addition to the design controls indicated above, there are a number of alarms and/or indications available to the operator to aid in detection of the potential for an overpressure transient and to aid in terminating the transient. These alarms and/or indications include such items as pressurizer high level alarms and high level indicators, pump status indicators and pump actuation alarms. Florida Power Corporation has also identified the adminstrative controls that are included in the plant operating procedures to reduce the potential for overpressurization.

Florida Power Corporation is presently engaged with Babcock and Wilcox to determine the long-term modifications that may be necessary and the analyses required to determine the most appropriate course of action for the long-term resolution of this matter.

We will review the results of the analyses and the proposed long-term modifications, when submitted, and will conclude upon an implementation plan upon completion of our review. In the interim we conclude that Crystal River Unit 3 can be permitted to operate under existing safety criteria based on the very large safety margins for unirradiated reactor vessels and under the current administrative procedures and overpressure design controls that minimize the potential for overpressurization.

5.5 Inservice Inspection Program

In Supplement No. 2 to our Safety Evaluation Report we stated that we were reviewing a request by Florida Power Corporation for written relief from the surveillance requirements for ASME Code Class 1, 2 and 3 components conforming to Section XI of the ASME Boiler and Pressure Vessel Code that is specified in Section 4.0.5 of the plant Technical Specifications. We stated that in the interim, Florida Power Corporation would perform those portions of the surveillance program in accordance with the plant Technical Specifications that are scheduled to be performed while operating in the refueling mode of operation and the cold shutdown mode of operation (Operational Modes 5 and 6). Florida Power Corporation has indicated to us that additional time will be needed in order to identify the specific items of exception to the Code and to bring the balance of the surveillance program into compliance with the Commission's regulations. In the interim Florida Power Corporation has identified the specific surveillance program that it will conduct for ASME Code Class 1, 2 and 3 components.

We have reviewed Florida Power Corporation's interim surveillance program for ASME Cod Class 1, 2 and 3 components. Based on our review we conclude that this surveillance program is acceptable as an interim program. Therefore, we will grant written relief the requirements of Section XI of the ASME Boiler and Pressure Vessel Code in accordance with Section 4.0.5 of the plant Technical Specifications for the period of plant operation from the date of issuance of the operating license (December 3, 1976) to March 3, 1977, a period of 90 days.

During this 90-day period, Florida Power Corporation will provide for our review the specific exceptions from the ASME Code requirements determined to be impractical, and the information to support these exceptions. We will report our evaluation of our review in a future supplement to this Safety Evaluation Report.

6.0 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.2 Containment Heat Removal Systems

In Section 6.3.3 of Supplement No. 2 to our Safety Evaluation Report we stated that we would report the results of our review of the modifications to the containment spray system. With regard to the modifications, Florida Power Corporation has informed us in its letter dated December 10, 1976 that the modifications to control the containment spray pump discharge have been completed and preoperational tests have been completed.

Based on our review of the modifications and the results of the preoperational tests we conclude that the modifications will preclude pump runout conditions and will prevent exceeding the margin available for net positive suction head and are, therefore, acceptable. We also conclude that manual throttling of the pump discharge valves during the recirculation phase of operation is acceptable.

The containment spray system is also used for injecting sodium thiosulfate into the spray water for iodine removal from the containment atmosphere following a loss-of-coolant accident and for injecting sodium hydroxide into the spray water for pH adjustment. During our review of the chemical addition to the spray water we had determined that the system, as originally proposed, was not acceptable because sufficient chemical addition would not occur during the injection phase, and no provisions had been made to continue the addition of chemicals during the recirculation phase. Florida Power Corporation modified the system and procedures to permit the continued addition of sodium hydroxide to the spray water during the recirculation phase of operation. This modification corrected one of the deficiencies of the chemical addition system, i.e., the addition of sufficient sodium hydroxide to maintain an acceptable pH. However, the modification did not correct the problem related to addition of the proper amount of sodium thiosulfate to assure that one percent by weight would be available in the sump.

We informed Florida Power Corporation of the remaining problem associated with the sodium thiosulfate addition. We are concerned that undesirable effects may occur outside the range of available test data when sodium thiosulfate is added to the system, unless the sodium thiosulfate is added in the proper quantity to give a one percent by weight solution in the containment sump. The chemical addition system and procedures did not assure that this quantity of sodium thiosulfate will be added following a postulated loss-of-coolant accident.

In our discussions with Florida Power Corporation we considered the elimination of sodium thiosulfate and relying on sodium hydroxide for iodine removal. Our experience in this regard is that iodine can be effectively removed from the containment atmosphere and retained in the water in the containment sump by the use of sodium hydroxide provided that the pH of the containment spray and containment sump is maintained within a range of 8.5 to 11.0. Florida Power Corporation has performed an analysis of the chemical additive system using sodium hydroxide only and has confirmed that the pH can be maintained within the acceptable range of 8.5 to 11.0

We have recalculated the iodine removal effectiveness of the system using sodium hydroxide alone and have confirmed that the dose guidelines of 10 CFR Part 100 will not be exceeded for this facility using sodium hydroxide only. The calculated dose exposures for the postulated loss-of-coolant accident are discussed in Section 15.0 of this Supplement.

On this basis we conclude that the chemical additive system using sodium hydroxide only is acceptable. We will, therefore, specify in the operating license that the sodium thiosulfate tank shall be isolated from the system by locking closed the valves in the tank discharge lines. Florida Power Corporation will provide a permanent modification to this system which will be installed prior to or during the first refueling outage.

6.3 Emergency Core Cooling System

6.3.3 Performance Evaluation

In Supplement No. 2 to our Safety Evaluation Report we stated that we would complete our review of the performance evaluation of the emergency core cooling system. We also stated that we would report the results of our review of the modifications to the containment spray system that are intended to preclude pump runout conditions. Our evaluation of the modifications to the containment spray system is discussed in Section 6.2.2 of this Supplement.

With regard to the performance evaluation of the emergency core cooling system, Florida Power Corporation has incorporated Babcock and Wilcox reports "B&W's ECCS Evaluation Model," BAW-10104, May 1975, and "ECCS Analysis of B&W's 177-FA Lowered-Loop NSS," BAW-10103, June 1975, into the Final Safety Analysis Report for Crystal River Unit No. 3. These reports were submitted pursuant to the requirements of Section 50.46 of 10 CFR Part 50 to demonstrate compliance with the acceptance criteria for the emergency core cooling system for the nuclear plants which utilize the 177 fuel assemblies with lowered loops. The bases for our acceptance of the principal portions of the Babcock and Wilcox evaluation model were set forth in the NRC staff's "Status Report by the Directorate of Licensing in the Matter of Babcock and Wilcox ECCS Evaluation Model Conformance to 10 CFR 50, Appendix K," dated October 1974 and Supplement 1 to this report dated November 13, 1974.

Subsequent to our evaluation as set forth in the NRC staff reports identified above, we determined that the method used by Babcock and Wilcox for calculating the fuel cladding temperature during the blowdown phase of the postulated loss-of-coolant accident does not conform to the requirements of Appendix K to 10 CFR Part 50 because the model allows for a return to nucleate boiling after critical heat flux conditions have been reached.

Babcock and Wilcox is presently considering several approaches to resolving this matter related to the return to nucleate boiling. In the interim until this matter is resolved, we have concluded that power operation of the facility is acceptable provided that the linear heat generation rate of the fuel elements is reduced by 20 percent of the values used in the performance evaluation of the emergency core cooling system. This limitation will be achieved by revising the limiting condition for operation as specified in Section 3.1.3.6 and 3.2.1 of the plant Technicial Specifications relative to regulating rod insertion limits and axial power imbalance, respectively. ;hese revisions to the Technical Specifications shall remain in effective until this matter is resolved to the satisfaction of the NRC staff. The limitations will more than compensate for the effect on peak clad temperature in the evaluation model that may exist because of the return to nucleate boiling heat transfer and is, therefore, conservative.

The Babcock and Wilcox report BAW-10103 presents analyses of a spectrum of break sizes, locations and configurations that are appropriate to Crystal Power Unit No. 3. These analyses identified the worst break as the 8.55 square foot double-ended break at the reactor coolant pump discharge. The maximum core-wide metal-water reaction was calculated to be 0.557 percent, which is below the allowable limit of one percent. The calculated values for the peak clad temperature and the local metal-water reaction were 2146 degrees Fahrenheit and 5.46 percent, respectively. These values are below the maximum allowable values of 2200 degrees Fahrenheit and 17 percent. The analyses also shows that the core geometry remains amenable to cooling and that long-term cooling can be established.

With regard to the break spectra, our guidelines require that a transition break and a sufficient number of small breaks be examined. In letters to Babcock and Wilcox dated February 4, 1976 and March 22, 1976 we informed Babcock and Wilcox of the need to submit analyses for the following break sizes:

(1) 0.04 square foot break using the small break model,

(2) a transition break using the large break model and the small break model, and

(3) a core flooding tank line break using the small break model.

Babcock and Wilcox has submitted preliminary information in regard to these break sizes which we are currently reviewing. In the interim until our review is complete, we conclude that power operation is acceptable since the linear heat generation rate of the fuel elements will be reduced by the reduction of regulating rod insertion limits and axial power imbalance as described above for the matter related to the return to nucleate boiling.

With regard to the single failure criterion, Appendix K to 10 CFR Part 50 requires that the combination of emergency core cooling subsystems to be assumed operative shall be those available after the most damaging single failure of equipment has occurred. In its report Babcock and Wilcox has conservatively assumed that all containment cooling systems operate to minimize the containment pressure, and has also assumed independently the loss of one diesel generator to minimize core cooling. In our status report we concluded that the application of the single failure criterion was to be confirmed during subsequent reviews of specific plants. Florida Power Corporation has confirmed that no single active failure at Crystal River Unit No. 3 would more severely degrade the emergency core cooling system than the assumptions indicated in the Babcock and Wilcox report as stated above.

We have conducted a review of the piping and instrumentation diagrams for Crystal River Unit No. 3 and the electrical schematic diagrams for the motor-operated valves in the emergency core cooling system. As a consequence of our review Florida Power Corporation modified the system to include automatic flow controllers in the discharge of the decay heat pumps to preclude the need for operator action to control pump runout which could result in insufficient net positive suction head for the pumps when the operator shifts to the long-term recirculation mode of operation following a postulated loss-of-coolant accident.

While we concluded that the modification to include automatic flow controllers as described above is acceptable, we determined that the operating range of the controlle (3000-3300 gallons per minute) must be narrowed to provide an adequate margin for net positive suction head. Florida Power Corporation proposed adjusting the operating range to 2800-3100 gallons per minute in order to assure an adequate margin. We have reviewed the acceptability of the operating range and concur that the proposed range is acceptable. The basis for our conclusion is that this range is within the flow range used in the vendor's evaluation model. Florida Power Corporation has performed preoperational tests to confirm that the flow controllers can maintain this flow range of 2800-3100 gallons per minute.

Also during our review of the instrumentation diagrams we determined that the low-wate level indication for the borated water storage tank did not meet all of the requirements for safety-related instrumentation. This indication is needed by the plant operator in order to allow the operator to determine when to shift to the long-term recirculation mode of core cooling following a postulated loss-of-coolant accident. Florida Power Corporation has stated in a letter dated December 10, 1976 that the level indication system will be modified to meet our requirements as discussed further in Section 7.3 of this Supplement. We have determined that this modification can reasonably be left for completion at a later date (within six months) on the basis of the very low likelihood of occurrence of the events which would concurrently lead to the need for this indication and the loss of function of the indication. Furthermore, since the drawdown time of the borated water storage tank can be determined from the measured pump discharge flows, the operator can estimate the time to reach the low water level when the shift must be made.

With regard to the containment pressure calculations, we concluded in our status report of the vendor's evaluation model that the containment pressure calculational model was acceptable, and that justification of plant dependent input parameters used in the containment pressure analysis would be submitted for our review of each plant. Florida Power Corporation has submitted justification for the plant dependent input data in a letter dated October 15, 1975. This justification allows us to compare the actual containment parameters for Crystal River Unit No. 3 with those assumed in the vendor's evaluation model.

Florida Power Corporation has evaluated the containment net-free volume, passive heat sinks, and operation of the containment heat removal systems with regard to the conservatism for the emergency core cooling system performance analysis. The heat removal systems were assumed to operate at their maximum capacities, and minimum values were assumed for the containment spray water and service water temperatures. The containment pressure analysis used in the vendor's evaluation model was thus demonstrated to be conservative for Crystal River Unit No. 3.

On the basis of our review of the containment pressure analysis we conclude that the plant dependent information used in the analysis for Crystal River Unit No. 3 is conservative and, therefore, the calculated containment pressures are in accordance with the requirements of Appendix K to 10 CFR Part 50.

We have reviewed the procedures for preventing excessive boric acid buildup in the reactor vessel during the long-term cooling period following a postulated loss-of-coolant accident. Florida Power Corporation will implement procedures which will allow adequate dilution of boron during the long-term cooling and which comply with the single failure criterion. These procedures will use a reactor coolant system hot leg drain line and hot leg injection line. The hot leg drain will direct reactor coolant from the hot leg, down the decay heat line, and back through the containment sump suction line to the containment sump. Water from the sump will than be pumped back to the reactor vessel using the second emergency core cooling train. In the event that a single active component failure does not allow operation of the hot leg drain mode, the operator has an alternative of selecting the hot leg injection mode to provide boron dilution. This alternative procedure uses water from the containment sump to the hot leg through the decay heat pump to provide dilution of the water in the upper plenum of the reactor pressure vessel.

In a letter to us dated December 10, 1976 Florida Power Corporation has informed us that tests have been conducted to verify that a minimum flow of 40 gallons per minute can be provided through the decay heat line to accomplish the necessary dilution of the boron. The 40 gallons per minute flow is the minimum flow that will maintain acceptable boron concentration in the reactor pressure vessel during long-term cooling following a postulated loss-of-coolant accident. The flow test verification using the decay heat line was accomplished by measuring the change in containment sump level. We require, however, that flow measurement instrumentation be installed to enable the operator to verify that the flow rate is at least 40 gallons per minute. Florida Power Corporation has agreed to install the flow instrumentation and to conduct tests to verify the system performance. We require that the system flow measurement system be installed and tested within six months and at the time when modifications to the level indication of the borated water storage tank are also accomplished.

Florida Power Corporation has conducted a review of equipment arrangement to determine if any components inside the containment will become submerged following a postulated loss-of-coolant accident. Based on this review, no equipment that is essential to the performance of the emergency core cooling system was identified that would be flooded.

Florida Power Corporation has submitted an analysis for operation with one reactor coolant pump idle (three pumps operating) by reference to the BAW-10103 report. This analysis was performed using a reduced power level of 77 percent of rated power and assuming the worse case break which is the 8.55 square foot, double-ended pipe rupture The worse break selected was located in the active leg of the partially idle loop. This break location yields the most degraded flow through the core during the first half of the blowdown and results in higher cladding temperatures. The maximum clad temperature calculated was 1766 degrees Fahrenheit. As a consequence of our review, a new analysis was submitted to reflect a more appropriate value of initial pressure in the fuel rods. The original analysis used an initial pressure of 1600 pounds per square inch, whereas the worse pressure should have been 760 pounds per square inch. The maximum cladding temperature for the reanalysis was 1784 degrees Fahrenheit. We conclude that the reanalysis acceptably supports operation of the reactor with one idle reactor coolant pump.

Since an analysis of the cooling performance of the emergency core cooling system with one idle pump in each reactor coolant loop has not been submitted for our review, power operation in this pump configuration will be prohibited by a condition to the operating license. This prohibition shall remain in effect until the analysis has been submitted to us and found to be acceptable. Single loop operation, i.e., operation with two idle pumps in one loop, is not allowed by plant Technical Specifications while the plant is operating in the power operation mode or the startup mode (Modes 1 and 2 as defined in Table 1.1 of the plant Technical Specifications).

Based on our review of the performance evaluation of the emergency core cooling system and in recognition of the modifications and preoperational tests that will be conducted as discussed above, we conclude that the system performance is in accordance with Appendix K to 10 CFR Part 50, with the exception of the analysis with regard to the return to nucleate boiling after critical heat flux conditions have been reached. Until this matter is resolved, and until we have completed our review of the additional small break analysis and transition break analysis, we conclude that plant operation at power is acceptable with the limitations imposed by the plant Technical Specifications on regulating rod insertion limits and axial power imbalance which shall remain in effect until these matters are satisfactorily resolved.

7.0 INSTRUCTION AND CONTROLS

7.2 Reactor Protection System

In Supplement No. 2 to our Safety Evaluation Report we stated that we would report the results of our evaluation of the need to modify the reactor coolant system flow indication in regard to the common pressure sensing line to all four differential pressure transmitters. We have now determined that the system should be modified to reduce the susceptibility of the system to false flow indication in the event of a break, leak or plugging of either the high pressure or low pressure sensing line.

We have informed Florida Power Corporation of the need to modify the system to reduce the susceptibility to false flow indication. We will review the proposed modifications when Florida Power Corporation completes its assessment and determines what modifications can be made, and we will require that approved modifications be implemented during or prior to the first refueling outage.

We conclude that until this matter is satisfactorily resolved, the surveillance requirements imposed by the plant Technical Specifications on the reactor protection system instrumentation (Table 4.3-1) and on the reactor coolant system operational leakage (Section 4.4.6.2) provide an acceptable assurance that breaks or leaks in the sensing lines will be detected. We also conclude that for the interim period, plugging of the sensing lines is highly unlikely.

7.3 Engineered Safety Feature Systems

In Supplement No. 2 to our Safety Evaluation Report we stated that we would review the details of the proposed modification to the low-water level indication system for the borated water storage tank. In a letter dated December 10, 1976 Florida Power Corporation provided us with the details and the schedule for completion of the modification. We have reviewed this information and conclude that the schedule for completion of this modification within six months is reasonable and therefore is acceptable, and that the design criteria that will be used for the design of the modification are acceptable.

15.0 ACCIDENT ANALYSIS

15.1 General

In our Safety Evaluation Report which was issued on July 5, 1974, we presented the potential offsite doses due to design bases accidents. The potential doses due to a postulated loss-of-coolant accident were based on the use of sodium hydroxide and sodium thiosulfate chemical addition to the containment quench spray system. As discussed in Section 6.2.2 of this Supplement, the chemical addition system has been modified to eleminate the use of sodium thiosulfate, and to rely on sodium hydroxide only. Consequently, we have recalculated the potential offsite doses due to the postulated loss-of-coolant accident for the modified chemical addition. The recalculated potential doses are tabulated below. The recalculation was based on the relative concentration factors given in Section 2.3.4 of our Safety Evaluation Report. The relative concentration for the 0-2 hour time period used in these calculations was 2.2×10^{-4} seconds per cubic meter. As discussed in Supplement No. 2 to our Safety Evaluation Report, the most recent meteorological data yields a relative concentration value of 9.6×10^{-5} seconds per cubic meter. This is a reduction of 2.3 in the relative concentration. The potential doses tabulated below are, therefore, conservatively derived and are well below the guideline values specified in 10 CFR Part 100.

POTENTIAL OFFSITE DOSES DUE TO POSTULATED LOSS-OF-COOLANT ACCIDENT

Two-hour Exclusion Boundary (1340 meters)		Course of Accident Low Population Zone (8047 meters)	
Thyroid (REM)	Whole Body (REM)	Thyroid (REM)	Whole Body (REM)
133	3	25	<1

20.0 FINANCIAL QUALIFICATIONS

In Supplement No. 2 to our Safety Evaluation Report we concluded that, based on our review of the application including Amendment No. 46 and Supplement No. 1 to the application, Florida Power Corporation and eleven co-owners are financially qualified to operate the facility. On November 17, 1976 the Florida Power Corporation submitted Amendment No. 50 to the application. This amendment provides an annual update of financial information for the Florida Power Corporation in accordance with the requirements of Appendix C to 10 CFR Part 50. In Amendment No. 50 the licensees state that the annual financial statements for the eleven co-owners will be submitted when they are available according to the requirements of Appendix C to 10 CFR Part 50.

We have reviewed the updated financial information provided in Amendment No. 50 and Supplement No. 1 to the application, and based on our review we reaffirm our conclusion as stated in Supplement No. 1 to our Safety Evaluation Report that the licensees are financially qualified to operate the facility according to the Commissions's regulations.

22.0 CONCLUSIONS

Based on our evaluation of the application as set forth in our Safety Evaluation Report issued on July 5, 1974, and Supplement Numbers 1 and 2 to that report issued on January 13, 1975 and December 3, 1976, respectively, and our evaluation as set forth in this Supplement, we have concluded that the facility operating license, DPR-72, issued on December 3, 1976 can be amended to allow power operation under the limiting conditions of operation as specified in the plant Technical Specifications, as amended, and as described in Section 3.8.1 and in Section 6.3.3 of this Supplement.

We conclude that the activities authorized by the amended operating license can be conducted without endangering the health and safety of the public, and we reaffirm our conclusions as otherwise stated in our Safety Evaluation Report.

APPENDIX A

CONTINUATION OF CHRONOLOGY OF RADIOLOGICAL REVIEW

CRYSTAL RIVER UNIT NO. 3

249. November 5, 1976	Letter from Florida Power Corporation regarding material surveillance program.
250. November 12, 1976	Letter from Florida Power Corporation advising of a proposed change to the plant Technical Specifications.
251. November 15, 1976	Letter from Florida Power Corporation containing additional information regarding the material surveillance program.
252. November 17, 1976	Submittal of Amendment No. 50 and Supplement No. 1 to application.
253. November 17, 1976	Letter from Florida Power Corporation regarding modification to the containment spray system.
254. November 18, 1976	Letter from Florida Power Corporation regarding proposed modifications to the level indication system for the borated water storage tank.
255. December 2, 1976	Letter from Florida Power Corporation regarding procedures and design controls to preclude overpressurization of the reactor coolant system.
256. December 3, 1976	Issuance of Supplement No. 2 to the Safety Evaluation Report.
257. December 9, 1976	Facility Operating License No. DPR-72 issued authorizing plant operation in refueling mode and cold shutdown mode.
258. December 9, 1976	Letter from Florida Power Corporation regarding status of information for review by NRC to authorize full power operation.

259. December 9, 1976 — Letter from Florida Power Corporation regarding proposed amendment to FSAR concerning the containment purging for hydrogen concentration.

260. December 10, 1976 — Letter from Florida Power Corporation regarding information supplied by B&W concerning small break analyses.

261. December 10, 1976 — Letter from Florida Power Corporation regarding modifications to the containment spray system.

262. December 10, 1976 — Letter from Florida Power Corporation regarding tests and modifications to ECCS to preclude boron concentration.

263. December 10, 1976 — Letter from Florida Power Corporation regarding proposed date for submittal of additional and revised data for ECCS Appendix K performance concerning return to nucleate boiling.

264. December 10, 1976 — Letter from Florida Power Corporation containing additional information regarding the modification to the water level indication system for the borated water storage tank.

265. December 9, 1976 — Submittal of report related to containment integrated leak rate test.

266. December 10, 1976 — Submittal of final report related to the containment dome delamination.

267. December 15, 1976 — Letter from Florida Power Corporation containing additional information related to proposed inservice inspection and testing of ASME Code Class 1, 2, and 3 components.

268. December 16, 1976 — Letter from Florida Power Corporation related to ECCS flow testing.

269. December 20, 1976 — Letter to Florida Power Corporation regarding analysis of ECCS.

270. December 28, 1976 — Letter to Florida Power Corporation regarding acceptability of seal material in hydraulic snubbers.