December 3, 1976

SUPPLEMENT NO. 2

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 in the matter of the application by Florida Power Corporation to operate the Crystal River Unit No. 3 facility was issued on July 5, 1974. Supplement No. 1 to the Safety Evaluation Report was issued on January 13, 1975. We indicated in Supplement No. 1 that there were a number of outstanding issues which required completion. Some of these issues related to technical areas under review by the Commission's staff at the time that Supplement No. 1 was issued. Other issues related to technical areas which required additional information from the applicant to permit us to confirm that certain requirements would be met by the applicant prior to issuing an operating license.

The purpose of this Supplement is to update our Safety Evaluation Report, and Supplement No. 1, by providing (1) our evaluation of additional information submitted by the applicant since the issuance of Supplement No. 1, (2) the results of our review of those technical areas for which our review was not completed at the time of issuance of Supplement No. 1, and (3) our evaluation of several new technical matters that have been identified since the issuance of Supplement No. 1.

Each of the sections in this Supplement is numbered the same as the section of the Safety Evaluation Report, and Supplement No. 1, that is being updated, and is supplementary to but not in lieu of the discussion in the Safety Evaluation Report and Supplement No. 1. Appendix A is a continuation of the chronology of our principal actions related to the processing of the application.

On July 31, 1975, the Florida Power Corporation concluded a sale for ten percent undivided ownership interest in the facility with eleven co-owners, ten of which are municipals and one of which is a cooperative. In accordance with the Commission's regulations the Florida Power Corporation, on behalf of itself and these eleven co-owners, filed an amendment to its application to admit these co-owners as joint applicants.

The new technical matters which have been identified since the issuance of Supplement No. 1, and the sections of this report where these matters are discussed, are as follows:

(1) The delamination of the reactor containment building dome that occurred during post-tensioning of the dome (Section 3.8.1).

(2) Asymetric loadings on the reactor pressure vessel supports which result from a postulated rupture of the reactor coolant piping at the cold leg nozzle (Section 3.9.4).

(3) Damage to a fuel assembly that occurred during receipt and unloading of new fuel and the replacement of the damaged assembly (Section 4.2.1).

(4) Potential pressurization of the reactor pressure vessel beyond the temperature-pressure limits specified in plant Technical Specifications (Section 5.2.1).

(5) Evaluation of the inservice inspection program to determine compliance with the Commission's regulations in Section 50.55a(g) of 10 CFR Part 50 as published in the FEDERAL REGISTER on February 12, 1976 (Section 5.5).

(6) Evaluation of the potential for pump runout conditions of the reactor building spray system pumps (Section 6.3.3).

(7) Requirement to modify the pressure sensing lines to the differential pressure transmitters of the reactor coolant flow indication to the reactor protection system (Section 7.2).

(8) Requirement to provide redundancy in the low-water level indication system for the borated water storage tank (Section 7.3).

(9) Reevaluation of the radioactive waste management systems to determine compliance with the Commission's regulations published in Appendix I to 10 CFR Part 50 (Section 11.0).

(10) Evaluation of the financial qualifications of the eleven additional co-owners of the facility and the evaluation of the updated finacial qualification of Florida Power Corporation (Section 20.0).

Based on our review of the application to date, we conclude that the Crystal River Unit 3 facility may load fuel and may be operated in the refueling mode of operation and the cold shutdown mode of operation (Modes 5 and 6 as defined in the Technical Specifications). Consequently, an operating license may be issued limiting plant operation to Operational Modes 5 and 6.

Before authorizing Crystal River Unit No. 3 to operate at a power level above criticality, such completeness of construction as is required for safe operation at the authorized power level must be verified by the Commission's Office of Inspection and Enforcement. We conclude that subject to the satisfactory completion of construction, including preoperational testing and procedures for operation, as verified by the Office of Inspection and Enforcement, the issuance of an operating license, which limits plant operation to Operational Modes 5 and 6, will not be inimical to the common defense and security, or to the health and safety of the public, and that upon favorable evaluation in a forthcoming Supplement to the Safety Evaluation Report of the outstanding safety items identified in this Supplement, the limitation on plant operation may be removed.

2.0 SITE CHARACTERISTICS

2.3 Meteorology

In Supplement No. 1 we stated that we would require the applicant to submit additional meteorological data to further verify the atmospheric dispersion conditions prior to the issuance of an operating license. This additional data would be for one year using a meteorological measurements program that is fully in accordance with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs."

In compliance with our requirement as stated above, the applicant has submitted one full year (January through December 1975) of meteorological data obtained from an onsite measurement program that we found to be in conformance with the recommendations of Regulatory Guide 1.23. Our evaluation of the onsite program is reported in Supplement No. 1.

We have reviewed the one full year of data, and have calculated the relative concentration (X/Q) values using joint frequency distributions of wind speed and wind duration by atmospheric stability for this one year period of January through December 1975. Wind speed and direction were measured at the 10-meter (33-foot) level, and atmospheric stability was determined by the vertical gradient between the 10-meter and the 53.3-meter (175-foot) levels. Data recovery for the joint frequency measurements was about 93 percent.

In the calculation of the short-term accidental releases from buildings and vents, we assumed a ground level release with a building wake factor of 1015 square meters. This factor was increased from the value of 925 square meters used in the evaluation presented in our Safety Evaluation Report issued on July 5, 1974, as a consequence of the applicant's reevaluation of the building wake factor.

The relative concentration value for the 0-2 hour time period for onshore air flow conditions (defined as those winds blowing from the southeast clockwise through the north directions) which is exceeded 5 percent of the time was calculated to be 9.6×10^{-5} seconds per cubic meter at the exclusion distance of 1340 meters. This value is a factor of 2.3 lower than the value of 2.2×10^{-4} seconds per cubic meter that was presented in our Safety Evaluation Report. Similar reductions were obtained for calculations at the low population zone distance.

Based on our review of the additional meteorological data and our evaluation as stated above, we conclude that the relative concentration estimates presented in Section 2.3.4 of our Safety Evaluation Report are higher compared to calculations based on the recent data collected at the site and are, therefore, conservative.

3.0 DESIGN CRITERIA-STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.8 Design of Category I (Seismic) Structures

3.8.1 Containment

On April 14, 1976, while placing cinch anchors in the dome concrete of the reactor building, the applicant's electrician found that the anchors would not hold. Subsequent soundings, core bearings and cutting investigations by the applicant indicated that the dome had delaminated into two layers. The approximate maximum thickness of the delaminated concrete was found to be 15 inches with a maximum gap of approximately two inches between the two layers. The plane area of delaminated concrete is approximately circular in shape with a 105 foot diameter. The delamination was not apparent via visual inspection of the dome surface. Additional cracks which scattered intermittently at various depths in the lower layer of the dome were found from the core borings.

On June 11, 1976 the applicant submitted an interim report "Reactor Building Dome Delamination." In the report, the applicant presented the original design criteria, the stresses and strains in concrete and steel liner of the original dome as well as the delaminated dome, and various investigations to determine the causes that led to the delamination. The applicant also proposed corrective action to be taken to assure that the containment structure, when so repaired will be capable of meeting the original design criteria as demonstrated by calculation and structural integrity tests.

We reviewed the report submitted on June 11, 1976, and had several meetings with the applicant regarding this matter. We stated our concerns in areas ranging from the determination of the causes of delamination to the corrective action to be taken. Even though the applicant did not identify positively any single or overriding mechanism as the cause of the delamination, we surmise from the facts presented by the applicant that:

(1) The characteristics of the dome concrete are such that it is crack-prone and localized cracks may have existed even before the prestressing force was applied.

(2) The coarse aggregates are fragile, and instead of acting as crack arrestors, they became the path of cracks.

(3) With the existence of precracks and the presence of fragile coarse aggregate, the radial tension accumulated from all sources and coupled with local stress concentration was so large that it overcame the much reduced tensile capacity of the concrete, resulting in the separation of the dome concrete.

After a series of discussions with the staff, the applicant concluded that the following procedure of repair would be the best from the standpoint of existing dome concrete integrity, cost, schedule and most importantly, of meeting our safety concerns:

(1) Holes were core-drilled into the lower concrete.
(2) The top delaminated concrete was removed.
(3) Final inspection of the 24-inch lower concrete was made.
(4) Lower level cracks were grouted with epoxy.
(5) Radial anchors were set and grouted in holes core-drilled in the lower concrete.
(6) New reinforcing steel was placed and concrete was cast to restore the dome to its original thickness.
(7) Eighteen tendons which were detensioned to study the effects on the structure were retensioned.

The dome thus repaired consists of the lower portion which is of prestressed concrete and the upper portion which is basically of reinforced concrete. With the dome thus repaired, the lower portion concrete is under higher prestress than the original design. For a specified concrete strength, this means that there will be greater loss of prestress. The dome design concrete strength was 5000 pounds per square inch, but the dome actual concrete strength was determined by the applicant to be about 6000 pounds per square inch. Therefore, higher concrete compressive stresses resulting from prestress would not likely produce any higher creep than that in the original dome.

Since there has not been any experience with the behavior of such a structure, we required the applicant to make a detailed analysis of the repaired dome and to have it instrumented so that a correlation between the predicted and measured behaviour can be established when the containment structure is subjected to a structural integrity test. On November 3, 1976 the structural integrity test of the Crystal River Unit No. 3 was completed. The results as provided in the applicant's November 4, 1976 preliminary test report indicate that the repaired dome behaved as expected.

We have reviewed the results presented by the applicant in the preliminary report, and based on that review, we conclude 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 will, however, review the applicant's final report to confirm our conclusion prior to authorizing power operation.

3.9 Mechanical Systems and Components

3.9.4 Analysis Methods Under Loss-of-Coolant Accident Loadings

We were informed on May 7, 1975 by a licensee of a pressurized water reactor, Virginia Electric and Power Company, that a loading due to a transient asymmetric pressure distribution over the reactor core barrel, resulting from a postulated pipe rupture at a particular location in the reactor coolant loop, had not been taken into account in the original design analysis of the reactor vessel support system for North Anna Units 1 and 2 (Docket Nos. 50-338 and 50-339). Similarly, the asymmetric loading from the transient differential pressures that would exist around the exterior of the reactor vessel from a postulated pipe rupture were not included in the original design analysis of Crystal River Unit No. 3. However, the symmetric loadings from such a pipe rupture were included in the original analysis of the reactor vessel support system.

It is our opinion that the question of the adequacy of reactor vessel support systems could be generic in nature and may apply to all pressurized water reactor facilities, especially those for which the design analyses were performed some time ago. We have therefore initiated a systematic review of this matter to determine what, if any, corrective measures may be required for specific facilities.

The results of studies reported to date indicate typically that, although the margins of safety may be less than originally intended, the reactor vessel support system will retain essential structural integrity and that the ultimate consequences of this postulated accident which could affect the general public are no worse than originally stated.

We have required that the applicant provide additional information for purposes of making the necessary reassessment of the reactor vessel supports for Crystal River Unit No. 3. This will include detailed analyses to determine the loads in the reactor vessel support system, evaluation of the full restraint capability of the support system, and computation of the safety margins of the support system.

After we have reviewed this information we will determine what modifications to Crystal River Unit No. 3, if any, are necessary to assure that acceptable margins of safety are maintained. If modifications are necessary we will require the applicant to make them.

Based on the results of our evaluation of this phenomenon to date and in recognition of the low probability of the particular pipe rupture which could lead to additional transient loads on the support system, we conclude that reactor operation will not create undue risk to the health and safety of the public and is therefore acceptable until we complete our generic review.

4.0 REACTOR

4.2 Mechanical Design

4.2.1 Fuel

On November 8, 1975, fuel assembly numbered 3A33 for the Crystal River Unit No. 3 facility was damaged during the receipt and unloading of new fuel on site. The applicant returned the damaged fuel assembly to the Babcock and Wilcox Company's Commercial Nuclear Fuel Plant for an assessment of required repairs or replacement. Upon inspection by Babcock and Wilcox Company, it was determined that the damaged fuel assembly would be replaced with a new assembly. The applicant submitted a report on March 26, 1976, which described the replacement fuel assembly. In response to our requests for additional information regarding the replacement fuel assembly, the applicant supplemented the report by a subsequent submittal on May 4, 1976. The report describes the replacement assembly along with analyses of the potential for cladding creep collapse, fuel/clad interaction, fuel densification and fuel swelling.

The replacement fuel assembly consists of 40 fuel rods removed from the original assembly and 168 replacement fuel rods. The only significant differences between the 168 replacement fuel rods and the original rods in the first fuel cycle loading relate to the internal spacers and the fuel pellets.

The replacement fuel rods contain spring spacers and Zircaloy tubular spacers compared to the corrugated tube spacers and zirconia ceramic spacers contained in the original rods. The newer type of spacers are representative of those currently in operation in several Babcock and Wilcox reactors which have been reviewed and determined to be acceptable and thus are acceptable for this reactor.

The fuel pellets in the replacement rods differ slightly in enrichment, density and active length as shown in Table 4.1. The potential effects of the low-density, 90.9 percent theoretical density (TD), pellets on thermal and mechanical performance were of particular concern to the NRC staff.

TABLE 4.1

COMPARISON OF FUEL PARAMETERS

FUEL ASSEMBLY	NO. FUEL RODS	ENRICHMENT W/% U_{235}	% THEORETICAL DENSITY	STACK LENGTH, IN.
3A33	40	1.93	92.5	144
	168	1.98	95.35	23-1/2 (Upper Zone)
		1.94	90.9	95-3/4 (Central Zone)
		1.98	95.35	23-1/2 (Lower Zone)
Remaining Assemblies in the First Core Load	208	1.93	92.5	144

Analyses of the thermal-hydraulic performance of the replacement fuel rods in replacement assembly 3A33 were performed by Babcock and Wilcox and reported in the March 26, 1976 report. A comparison of the results of these analyses with analyses of the 40 original fuel rods in assembly 3A33 and the fuel rods in the remaining 176 fuel assemblies is shown in Table 4.2. The results of the analyses show that, except for the engineering hot channel factor, all thermal-hydraulic performance parameters for the replacement fuel rods in fuel assembly 3A33 are not more restrictive than for the fuel rods in the limiting fuel assembly in the remaining 176 fuel assemblies.

TABLE 4.2

COMPARISON OF THERMAL-HYDRAULIC

PARAMETERS

THERMAL-HYDRAULIC CRITERIA	168 REPLACEMENT FUEL RODS IN FUEL ASSEMBLY 3A33	40 ORIGINAL FUEL RODS IN 3A33 AND FUEL RODS IN REMAINING 176 FUEL ASSEMBLIES
1. Linear Heat Rate Limit Based on Central Fuel Melting, KW/Ft.		
For Fuel Density:		
a. 95.35% Theoretical Density	21.46	-
b. 90.9% Theoretical Density	19.96	-
c. 92.5% Theoretical Density	-	19.7
2. Average Linear Heat Rate, KW/Ft.	5.765	5.771
3. Average Fuel Temperatures (Stored Energy), F		
a. At Average Linear Heat Rate:		
(1) 95.35% Theoretical Density	1285	-
(2) 90.9% Theoretical Density	1327	-
(3) 92.5% Theoretical Density	-	1335
b. At 18 KW/Ft.		
(1) 95.35% Theoretical Density	2840	-
(2) 90.9% Theoretical Density	3066	-
(3) 92.5% Theoretical Density	-	3110
4. Engineering Hot Channel Factor	1.026	1.014
5. DNBR Penalty Due to Fuel Densification, %	1.9	2.9

Cladding creep collapse analyses were performed by Babcock and Wilcox in accordance with material properties and design procedures set forth in Topical Report B&W-10084P-A, titled "Program to Determine In-Reactor Performance of B&W Fuels". The evaluation was completed using the creep ovalization analysis code (CROV) described in Section 3 of the cited report which was previously reviewed and approved by the NRC staff. In addition other conservatisms were introduced, as described in the May 4, 1976 report. Results of the analyses indicated a collapse time greater than 14,000 hours, compared to the required 10,320 hours associated with the single fuel cycle burn of assembly 3A33.

Pellet/cladding mechanical interaction and fuel swelling effects were addressed by Babcock and Wilcox in its cladding strain analysis. Of the pellet densities used in assembly 3A33, the 90.9% theoretical density pellets represent the limiting pellet/cladding mechanical interaction case at the peak pellet burnup seen by the assembly. Accordingly, cladding strain analyses were performed on the 90.9% theoretical density fuel corresponding to the worst-case specification dimensions and the as-built, 2-sigma dimensions. The analyses were performed in accordance with material data and design models set forth in Section 3 of Topical Report B&W-10054, Revision 2, titled "Fuel Densification Report". This represented the same approach as used in the Final Safety Analysis Report for Crystal River Unit No. 3 except that additional conservatisms were introduced for the 3A33 analyses. The results of the analyses indicated that the total circumferential strain resulting from pellet/cladding mechanical interaction for the worst-case-specification analysis and the as-built dimensions analysis were 0.80% and 0.48%, respectively, as compared to the Babcock and Wilcox design value of 1.42%.

The mechanical design and thermal analysis aspects of the 168 replacement fuel rods in replacement fuel assembly 3A33 have been analyzed by Babcock and Wilcox, using codes and methods previously reviewed and found acceptable by the NRC staff and in accordance with material data and design models approved by the NRC staff. Evaluations of the potential for cladding creep collapse, pellet/cladding mechanical interaction, fuel densification and fuel swelling were made. The results of these analyses have shown that the 3A33 fuel rods are within acceptable design limits for first fuel cycle operation. Based on our review of the results of the analyses of the 3A33 replacement fuel assembly which demonstrates the adequacy of the replacement assembly, the replacement assembly is acceptable with regard to the potential for cladding creep collapse, pellet/cladding mechanical interaction, fuel densification and fuel swelling.

With regard to the effects of the higher engineering hot channel factor of the replacement fuel rods compared to the original fuel rods in the replacement bundle, we have not completed our review of this item. We conclude that plant operation, which will be limited to the refueling mode and the cold shutdown mode of operation, is acceptable while we complete our review of the effects of the higher engineering hot channel factor. We will report the results of our review in a future supplement to our Safety Evaluation Report prior to authorizing power operation.

5.0 REACTOR COOLANT SYSTEM

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Fracture Toughness

Reactor Coolant System Overpressurization

Our review of recent operating experience indicated that there have been several reported incidents at pressurized water reactor facilities in which pressure-temperature limits established in accordance with Appendix G to 10 CFR Part 50 have been exceeded. Most of the incidents have occurred during reactor startup or shutdown while the reactor coolant system was in a solid-water condition.

Since operating experience has shown that these pressure transients have occurred at greater frequency than originally anticipated, the staff considers it prudent to take steps to reduce the likelihood of their occurrence. We have therefore initiated a systematic review to determine what corrective measures should be required at specific facilities. In August of 1976 we requested that all licensees of operating pressurized water reactor facilities evaluate their system design and determine its susceptibility to overpressurization events. We requested that the licensees then develop permanent measures to reduce the likelihood of such events. If the permanent measures could not be quickly implemented, we further requested that the licensees institute short-term measures to minimize the likelihood of pressure transient events during the interim period prior to installation of long-term corrective measures.

We have requested the applicant in our letter dated October 1, 1976, to describe the short-term procedures and equipment modifications to mitigate the problem during the operation of Crystal River Unit No. 3. We have also initiated discussions with reactor vendors, including Babcock and Wilcox, on a generic basis relative to overpressurization during the conditions described above. Crystal River Unit No. 3 and other Babcock and Wilcox plants contain features which reduce the probability of such an event occurring. We will continue our review of this matter and report any changes in status in a future supplement to our Safety Evaluation Report.

In the interim until such determinations have been made and measures taken to preclude or minimize the probability of overpressurization, we have concluded that a licensing action which limits plant operation to the refueling mode and cold shutdown mode is acceptable. The basis for our conclusion is that overpressurization is very unlikely while the plant is limited to the cold shutdown condition and the consequences of overpressurization will not result in an unsafe condition for a cold, subcritical and unirradiated core. Prior to removing the limitations on plant operation we will evaluate the measures to be taken by the applicant and we will report the results of our evaluation in a future supplement to our Safety Evaluation Report.

Materials Surveillance Program

In our Safety Evaluation Report we stated that the applicant's program for monitoring the effects of irradiation and temperature of the reactor beltline materials throughout the service life of the reactor pressure vessel was acceptable. However, in a letter to us dated September 28, 1976, the applicant indicated to us that installation of surveillance capsules would be deferred until the first refueling. Subsequently, the applicant reconsidered this matter and concluded that two capsules will be installed now for irradiation during the first fuel cycle with additional capsules to be installed at the first refueling as described in the applicant's letter dated November 5, 1976.

At our request the applicant provided additional information regarding the material surveillance program in a letter to us dated November 15, 1976. This information identified the number, type and material identification of the specimens in each of the capsules, and identified the location of the specimens in relationship to the fracture toughness requirements of Appendix G to 10 CFR Part 50 and Appendix G to the ASME Code. The information also identified the specimen withdrawal schedule as related to the requirements of Appendix H to 10 CFR Part 50.

We have reviewed the additional information provided by the applicant in regard to the surveillance program for the reactor vessel material. We have determined that the program is in compliance with the requirements of Appendices G and H to 10 CFR Part 50 to the extent practical.

Compliance with all the guidelines and provisions of Appendices G and H to 10 CFR Part 50 was not possible for the pressure retaining components of Crystal River Unit No. 3 since all of the required test material were not available because the Appendices G and H, which require this test material, were published in the FEDERAL REGISTER on July 17, 1973, and the construction permit for Crystal River Unit No. 3 was granted on September 25, 1968. The applicant was, therefore, not able to comply with that specific requirement.

Appendix G requires that test specimens for monitoring the reactor vessel beltline be taken from excess material and welds in the shell courses following completion of the longitudinal weld joint. In addition, a minimum value of the upper shelf energy of 75 foot-pounds is required. Justification for deviation from the guidelines and provisions of Appendix G has been presented for Crystal River Unit No. 3 in Babcock and Wilcox Topical Reports BAW-10046P, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G," and BAW-10100A, "Reactor Vessel Material Surveillance Program - Compliance with 10 CFR 50, Appendix H, for Oconee Class Reactors."

Appendix H requires in part that the surveillance specimens be selected from material adjacent to the fracture toughness test specimens required by Appendix G to 10 CFR Part 50. In lieu of the actual material, surveillance specimens have been selected on the basis of equivalent composition. This program is consistent with surveillance programs that have been found acceptable for other light water reactors.

Compliance to the fracture toughness test methods and procedures required by Appendices G and H to 10 CFR Part 50, and to Appendix G to the ASME Boiler and Pressure Vessel Code, 1971 Edition, including the Winter 1972 Addenda, to the extent practical, provide reasonable assurance that adequate safety margins exist against the possibility of nonductile behavior or rapidly propagating fracture for the pressure-retaining components of the reactor coolant pressure boundary.

The integrated surveillance program constitutes an acceptable basis for monitoring irradiation and temperature induced changes in the fracture toughness of the materials of the rector vessel beltline region, and satisfies the requirements of General Design Criterion 31 of Appendix A to 10 CFR Part 50.

Based on our review of the fracture toughness and material surveillance program and the determinations indicated above, we conclude that the program is acceptable.

5.5 Inservice Inspection Program

The plant Technical Specifications which will be included as Appendix A to the facility operating license, presently require that the inservice inspection and testing of ASME Code Class 1, 2 and 3 components shall conform to Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition, and Addenda through Winter 1975, except where specific written relief is granted by the Commission, for the period from issuance of the license to the start of commercial operation. For the time following the start of commercial operation, inservice inspection and testing shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by Section 50.55a(g) of 10 CFR Part 50 that was published in the FEDERAL REGISTER on February 12, 1976, except where written relief is granted by the Commission pursuant to the regulation.

The applicant has sought relief from certain requirements as stated above. We are currently reviewing whether relief can be granted from these requirements, and we will report the results of that review in a future supplement to our Safety Evaluation Report. Depending on the results of our review, we will, if necessary, revise the Technical Specifications by amendment to the operating license. In the interim, the applicant will be required to perform those portions of the inservice inspection and testing that are scheduled to be performed in accordance with the present Technical Specifications.

6.0 ENGINEERED SAFETY FEATURES

6.3 Emergency Core Cooling System

6.3.3 Performance Evaluation

In our Safety Evaluation Report, and in Supplement No. 1, we stated that we would evaluate the performance of the emergency core cooling system to determine its conformance to the requirements of Appendix K to 10 CFR Part 50. The applicant has submitted an evaluation of the emergency core cooling system as required by Section 50.46 of 10 CFR Part 50. We are currently reviewing this evaluation. In the interim until our review is completed, we conclude that plant operation, which is limited to the refueling mode and the cold shutdown mode of operations, is acceptable. The basis for our conclusion is that there is no accident that could occur under these conditions that would require operation of the emergency core cooling system.

Prior to removing the limitations on plant operation, we will complete our review of the performance evaluation of the emergency core cooling system, and we will report the results of our review in a future supplement to our Safety Evaluation Report.

During the course of our review, and subsequent to the issuance of Supplement No. 1 to our Safety Analysis Report, we determined that modifications may be necessary to preclude a runout condition on the containment spray pumps or to prevent exceeding the margin available for net positive suction head to the pumps. In response to our concern, the applicant modified the spray system by installing safety-related automatic flow switches which will prevent the conditions described above by throttling the pump discharge valves to maintain the required discharge flow.

The modification to the spray system consists of two separate sets of flow switches. One set will control flow during the injection phase of operations while the second set will control during the recirculation phase. During tests of the installed system, the second set of switches failed to perform the required function due to the inherent valve characteristics in the near closed position at the lower flow rate for recirculation phase of operation, which resulted in valve position oscillations. For this reason, the spray system is being modified to use manual throttling of the pump discharge valve during the recirculation phase of operation.

We will review these modifications to the spray system and will report the results of our review in a future supplement to our Safety Evaluation Report. In the interim until our review is completed, we conclude that plant operation in the refueling mode and cold shutdown mode of operation is acceptable. The basis for our conclusion is that there is no accident that could occur under these plant operation conditions that would require operation of the spray system.

7.0 INSTRUMENTATION AND CONTROLS

7.2 Reactor Protection System

During the course of our review, and subsequent to the issuance of Supplement No. 1 to our Safety Evaluation Report, we determined that modifications may be necessary to the pressure sensing lines of the reactor coolant system flow indication. All four differential pressure transmitters in each flow measuring device in each of two coolant hot legs are connected to a common line. Such a system of common connection is susceptible to false flow indication in the event of a leak, break or plugging of the high pressure sensing line or the low pressure sensing line.

We are currently evaluating this matter associated with the common pressure line to all differential pressure transmitters in order to determine the need for modifications. In the interim until we complete our evaluation, we conclude that plant operation in the refueling mode and the cold shutdown mode of operation is acceptable. The basis for our conclusion is that the measurement of reactor coolant flow is needed only during power operation. Prior to removing the limitations on plant operation we will complete our evaluation and report the results in a future supplement to our Safety Evaluation Report.

7.3 Engineered Safety Feature Systems

During the course of our review of the Crystal River Unit No. 3 facility, and subsequent to the issuance of Supplement No. 1 to our Safety Evaluation Report, we determined that the low-water level indication for the borated water storage tank should be modified to provide redundant indication in the control room. We informed the applicant of the need to modify the level indication and, in a letter to us dated November 18, 1976, the applicant made a commitment to meet our requirements.

The applicant will install two independent level indication channels powered by separate power supplies. The equipment will be seismically and environmentally qualified and will meet the requirements of IEEE-279 to the extent practicable. The applicant also agreed to provide us with the details of the modification and a schedule for installation of the modification.

We will review the details regarding the modification to the level indication of the borated water storage tank and will report the results of that review in a future supplement to our Safety Evaluation Report. In the interim until our review is complete, we find that this commitment is acceptable, and we conclude that the plant may safely be operated in the refueling mode and the cold shutdown mode of operations. The basis for our conclusion is that for a core in a cold, subcritical and unirradiated condition, no accident could occur which would require actuation of the emergency core cooling system.

11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 Introduction

On May 28, 1976, the applicant provided the necessary information to permit an evaluation with respect to Appendix I to 10 CFR Part 50. The applicant elected to show conformance with the Commission's September 4, 1975 amendment to Appendix I in lieu of performing a detailed cost-benefit analysis as required by Section II.D of Appendix I.

11.2 Evaluation

We have evaluated the radioactive waste management systems installed at Crystal River, Unit No. 3, to reduce the quantities of radioactive materials released to the environment in liquid and gaseous effluents. These systems have been previously described in Section 3.4 of the Final Environmental Statement, dated May 1973, and in Section 11.0 of our Safety Evaluation Report, dated July 5, 1974. Based on more recent operating data applicable to the Crystal River, Unit No. 3 facility, and on changes in our calculational model, we have generated new liquid and gaseous source terms to determine conformance with Appendix I. These values are different from those given in Tables 3.2 and 3.3 of the Final Environmental Statement.

The new source terms, shown in Tables 11.1 and 11.2, were calculated using the models and methodology described in NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code)," April 1976. These source terms were used to calculate the doses as described below. The dispersion of radionuclides in and the disposition of radionuclides from the atmosphere were based on analyses performed by the staff for this evaluation using the methodology provided in Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," March 1976.

The mathematical models used to perform the dose calculations are contained in Regulatory Guide 1.109, "Calculation of Annual Average Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Implementing Appendix I," March 1976.

Included in our analysis are dose evaluations of three effluent categories: (1) pathways associated with liquid effluent releases to the Gulf of Mexico, (2) noble gases released to the atmosphere, and (3) pathways associated with radioiodines, particulates, carbon-14 and tritium released to the atmosphere.

The dose evaluation of pathways associated with liquid effluents was based on the maximum exposed individual. The dietary and living habits for an adult individual included the consumption of 21 kilograms per year of fish harvested from the Gulf of Mexico in the immediate vicinity of the discharge, and recreational use of its shoreline for 10 hours per year. There are no drinking water sources receiving Crystal

TABLE 11.1

CALCULATED RELEASES OF RADIOACTIVE MATERIALS
IN LIQUID EFFLUENTS
FROM CRYSTAL RIVER, UNIT NO. 3

Nuclide	
Corrosion & Activation Products	
Chromium 51	1(-4)
Manganese 54	1(-3)
Iron 55	1.1(-4)
Iron 59	6(-5)
Cobalt 58	5(-3)
Cobalt 60	8.8(-3)
Zirconium 95	1.4(-3)
Niobium 95	2(-3)
Neptunium 239	5(-5)
Fission Products	
Bromine 83	1.1(-4)
Strontium 89	2(-5)
Strontium 91	3(-5)
Yttrium 91m	2(-5)
Molybdenum 99	3.4(-2)
Technetium 99m	3.4(-2)
Ruthenium 103	1.4(-4)
Ruthenium 106	2.4(-3)
Silver 110m	4.4(-4)
Tellurium 127m	1(-5)

Nuclide	
Fission Products (cont'd)	
Tellurium 127	5(-5)
Tellurium 129m	8(-5)
Tellurium 129	6(-5)
Iodine 130	1(-4)
Tellurium 131m	5(-5)
Iodine 131	7.5(-2)
Tellurium 132	9.5(-4)
Iodine 132	3(-3)
Iodine 133	1.8(-2)
Iodine 134	8(-5)
Cesium 134	1.6(-2)
Iodine 135	8.3(-3)
Cesium 136	1.1(-3)
Cesium 137	2.6(-2)
Barium 137m	2.3(-3)
Cerium 144	5.2(-3)
All others	6(-5)
Total (except tritium)	2.5(-1)
Tritium	500

Exponential notation; 1(-4) = 1×10^{-4}

Nuclides whose release rates are less than 10^{-5} Curies per year per reactor are not listed individually, but are included in the category "All Others."

TABLE 11.2

CALCULATED RELEASES OF RADIOACTIVE MATERIAL
IN GASEOUS EFFLUENTS FROM
CRYSTAL RIVER, UNIT NO. 3

CURIES PER YEAR PER REACTOR

Radionuclide[d]	Reactor Building	Auxiliary Building	Turbine Building	Air Ejector	Decay Tanks	Total
Krypton 83m	a	4	a	2	a	6
Krypton 85m	1	2(+1)	a	1.2(+1)	a	3.3(+1)
Krypton 85	9	3	a	2	2(+2)	2.1(+2)
Krypton 87	a	1.1(+1)	a	7	a	1.8(+1)
Krypton 88	1	3.6(+1)	a	2.3(+1)	a	6(+1)
Krypton 89	a	a	a	a	a	a
Xenon 131m	1.3(+1)	6	a	4	a	2.4(+1)
Xenon 133m	2(+1)	2.6(+1)	a	1.7(+1)	a	6.3(+1)
Xenon 133	2.2(+3)	1.5(+3)	a	9.5(+2)	a	4.7(+3)
Xenon 135m	a	2	a	2	a	4
Xenon 135	8	6(+1)	a	3.7(+1)	a	1(+2)
Xenon 137	a	2	a	1	a	3
Xenon 138	a	8	a	5	a	1.3(+1)
Iodine 131	1.3(-3)	5.5(-3)	1.1(-3)	3.5(-2)	a	4.3(-2)
Iodine 133	4.3(-4)	1.1(-2)	2.1(-3)	6.7(-2)	a	8.1(-2)
Manganese 54	2.2(-4)	1.8(-4)	c	c	4.5(-5)	4.4(-4)
Iron 59	7.5(-5)	6(-5)	c	c	1.5(-5)	1.5(-4)
Cobalt 58	7.5(-4)	6(-4)	c	c	1.5(-4)	1.5(-3)
Cobalt 60	3.4(-4)	2.7(-4)	c	c	7(-5)	6.8(-4)
Strontium 89	1.7(-5)	1.3(-5)	c	c	3.3(-6)	3.3(-5)
Strontium 90	3(-6)	2.4(-6)	c	c	6(-7)	6(-6)
Cesium 134	2.2(-4)	1.8(-4)	c	c	4.5(-5)	4.4(-4)
Cesium 137	3.8(-4)	3(-4)	c	c	7.5(-5)	7.5(-4)
Tritium	5.1(+2)	c	c	c	c	5.1(+2)
Carbon 14	1	a	a	a	7	8
Argon 41	2.5(+1)	c	c	c	c	2.5(+1)

a = less than 1.0 Curies per year per reactor for noble gases and carbon-14, less than 10^{-4} Curies per year per reactor for iodine

b = exponential notation; 1.4(-2) = 1.4×10^{-2}

c = less than 1 percent of total for this nuclide

d = radionuclides not listed are released in quantities less than those specified in notes a and c from all sources

River, Unit No. 3 liquid effluents. The maximum dose commitment resulting from exposure to water from the Gulf of Mexico was estimated to be 0.0027 millirem per year (total body) and 0.0026 millirem per year (Gastro Intestinal tract) for an adult.

The dose evaluation of noble gases estimated to be released to the atmosphere included a calculation of beta and gamma air doses at the site boundary and total body and skin doses at the residence having the highest dose. The maximum air doses at the site boundary were found at one mile northwest relative to Crystal River, Unit No. 3. The location of maximum total body and skin doses was determined to be at a residence three miles northeast of Crystal River, Unit No. 3.

The dose evaluation of pathways associated with radioiodine, particulates, carbon-14 and tritium estimated to be released to the atmosphere was also based on the maximum exposed individual. This individual is a child whose diet included the consumption of 250 kilograms per year of crops assumed to be produced at the location of the above mentioned residence, for lack of more detailed crop production information.

The assumption that all of the vegetables ingested by this child are produced at his residence is conservative. Thus, the estimated actual radiological dose to the thyroid of this individual will be lower.

As shown in Table 11.1, the expected quantity of radioactive materials released in liquid effluents from Crystal River, Unit No. 3 will be less than five Curies per year per reactor (0.25 Curies per year), excluding tritium and dissolved gases, in conformance with the amendment to Section II.D of Appendix I to 10 CFR Part 50. The liquid effluents estimated to be released from Crystal River, Unit No. 3, will not result in an annual dose or dose commitment to the total body or to any organ of an individual, in an unrestricted area from all pathways of exposure, in excess of five millirem, as shown in Table 11.3.

Based on our evaluation of the gaseous radwaste management systems, the total quantity of radioactive materials estimated to be released in gaseous effluents from the facility will not result in an annual gamma air dose in excess of 10 millirads and a beta air dose in excess of 20 millirads at every location near ground level, at or beyond the site boundary, which could be occupied by individuals as shown in Table 11.3. As shown in Table 11.2, the annual total quantity of iodine-131 released in gaseous effluents will be less than one Curie per year per reactor (0.043 Curie per year) in conformance with the September 4, 1975 amendment to Appendix I. The annual total quantity of radioiodine and radioactive particulates estimated to be released in gaseous effluents from the facility will not result in an annual dose or dose commitment to any organ of an individual in an unrestricted area from all pathways of exposure in excess of 15 millirem, as shown in Table 11.3.

TABLE 11.3

COMPARISON OF CRYSTAL RIVER, UNIT NO. 3 WITH APPENDIX I TO 10 CFR PART 50, SECTIONS II.A, II.B AND II.C (MAY 5, 1975)[a] AND SECTION II.D, ANNEX (SEPTEMBER 4, 1975)[b]

Criterion	Appendix I[a] Design Objectives[c]	Annex[b] Design Objectives	Calculated Doses Unit No. 3
Liquid Effluents			
Dose to total body from all pathways (infant)	3 mrem/yr/unit	5 mrem/yr/site	0.0027 mrem/yr/unit
Dose to any organ from all pathways	10 mrem/yr/unit	5 mrem/yr/site	0.0026 mrem/yr/unit
Noble Gas Effluents[d]			
Gamma dose in air	10 mrad/yr/unit	10 mrad/yr/site	0.38 mrad/yr/unit
Beta dose in air	20 mrad/yr/unit	20 mrad/yr/site	0.84 mrad/yr/unit
Dose to total body of an individual	5 mrem/yr/unit	5 mrem/yr/site	0.45 mrem/yr/unit
Dose to skin of an individual	15 mrem/yr/unit	15 mrem/yr/site	0.95 mrem/yr/unit
Radioiodines and Other Radionclides Released to the Atmosphere[e]			
Dose to any organ from all pathways (child-thyroid)	15 mrem/yr/unit	15 mrem/yr/site	0.069 mrem/yr/unit

[a] Federal Register, V. 40, p. 19442, May 5, 1975.

[b] Federal Register, V. 40, p. 40816, September 4, 1975.

[c] Design Objectives given on a site basis. Therefore, these design objectives apply to 1 unit at the site.

[d] Limited to noble gases only.

[e] Carbon-14 and Tritium have been added to this category.

mrem/yr/unit = millirem per year per unit
mrad/yr/unit = millirad per year per unit
mrem/yr/site = millirem per year per site
mrad/yr/site = millirad per year per site

11.3 Conclusion

Based on our evaluation, the radwaste treatment systems for Crystal River, Unit No. 3, are capable of maintaining releases of radioactive materials in liquid and gaseous effluents during normal operation such that the doses will not exceed the design objectives of Sections II.A, B and C of Appendix I of 10 CFR Part 50.

Our evaluation also shows that the applicant's design of Crystal River, Unit No. 3, satisfies the design objectives set forth in the Commission's Concluding Statement of Position of the Regulatory Staff (and its Attachment) - Public Rulemaking Hearing on: Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criteria "As Low As Practicable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactors, Docket Number RM-50-2, Washington, D.C., February 20, 1974, and specified in the option provided by the Commission's September 4, 1975 amendment to Appendix I and, therefore, meets the requirements of Section II.D of Appendix I of 10 CFR Part 50.

We conclude that the liquid and gaseous radwaste treatment systems will reduce radioactive materials in effluents to "as low as is reasonably achievable levels" in accordance with 10 CFR Part 50.34a and, therefore, are acceptable.

13.0 CONDUCT OF OPERATIONS

13.5 Industrial Security

The applicant has revised the Industrial Security Plan for the protection of the facility from industrial sabotage. These revisions are identified as Revisions 1, 2 and 3 submitted to the Commission by letters dated December 21, 1973, February 18, 1976, and March 19, 1976, respectively.

We have reviewed the revisions to the Industrial Security Plan and have determined that these revisions do not decrease the provisions for industrial security previously reviewed and found to be acceptable as stated in our Safety Evaluation Report. On this basis we confirm our conclusion that the Industrial Security Plan is acceptable.

20.0 FINANCIAL QUALIFICATIONS

20.1 Introduction

The Nuclear Regulatory Commission's regulations relating to financial data and information required to establish financial qualifications for applicants for plant operating licenses appear in Section 50.33(f) of 10 CFR Part 50 and Appendix C to 10 CFR Part 50. In accordance with these regulations the Florida Power Corporation, on behalf of itself and eleven additional co-owners (listed in Table 20.1), has filed Amendment No. 46 and Supplement No. 1 to its application for licenses for Crystal River Unit 3 Nuclear Generating Plant. Amendment No. 46 is Florida Power Corporation's annual update of its application for licenses. Supplement No. 1 consists primarily of financial information on the co-owners that was requested by the staff for use in completing its evaluation on financial qualifications. The following analysis summarizes our review of the application including the amendments and Supplement No. 1 and addresses the financial qualifications of the Florida Power Corporation and the eleven additional co-owners (ten of which are municipals and one of which is a cooperative) to operate the subject facility and, if necessary, to permanently shut it down and maintain it in a safe condition.

On July 31, 1975, the Florida Power Corporation concluded a sale for an aggregate ten percent undivided ownership interest in the facility with the eleven additional co-owners. Table 20.1 indicates each co-owner's percentage entitlement in the electricity capacity and output of the plant. These percentages are identical to each co-owner's ownership percentage in the facility.

20.2 Estimated Operating And Shutdown Costs

For the purpose of estimating the plant annual operating costs, the Florida Power Corporation assumed that the unit would begin operation in September 1976. Florida Power Corporation's estimate of the annual cost of operating the plant during each of the first five years of commercial operation are presented in Table 20.2. The unit costs (mills per kwh) are based on a unit capacity of 855 MWe and on the following projected plant capacity factors: 1976-70%; 1977-86%; 1979-90%; and 1980-92%. The five-year average costs were calculated by converting the September thru December costs of 1976 into an annual cost in combination with the annual estimates for 1977 through 1980.

TABLE 20.1

LIST OF CO-OWNERS AND PERCENTAGE ENTITLEMENT IN ELECTRICAL CAPACITY AND OUTPUT

Participant	Percent Entitlement
Florida Power Corporation	90.0000
City of Alachua, Fla.	0.0779
City of Bushnell, Fla.	0.0388
City of Gainesville, Fla.	1.4079
City of Kissimmee, Fla.	0.6754
City of Leesburg, Fla.	0.8244
City of New Smyrna Beach, Fla. and Utilities Commission, New Smyrna Beach, Fla.	0.5608
City of Ocala, Fla.	1.3333
Orlando Utilities Commission and City of Orlando, Fla.	1.6015
Sebring Utilities Commission	0.4473
Seminole Electric Cooperative, Inc.	1.6994
City of Tallahassee, Fla.	1.3333
TOTAL	100.0000

TABLE 20.2

ANNUAL COST OF COMMERCIAL OPERATION

	Total Cost (dollars in thousands)	Mills per Kwh
1976	$ 27,615	15.80
1977	102,240	15.87
1978	99,880	15.51
1979	96,594	14.33
1980	94,049	13.65
5-year average	95,122	15.03

The above estimated plant operating costs compare favorably with the Florida Power Corporation's recent revenue experience. For the 12-month period ended May 31, 1975, Florida Power Corporation's unit price was 34.10 mills per kwh on its system-wide sales of electric power to all customers, well above the total estimated unit operating costs for the subject facility. The source of funds from the participants to cover operating costs and shutdown and maintenance costs is discussed in Section 20.3, below.

In estimating the costs of permanently shutting down the facility and maintaining it in a safe shutdown condition, the Florida Power Corporation assumed that the reactor and its associated nuclear systems would be left in place and that all nuclear fuel would be removed from the plant and transported offsite for final processing. The Florida Power Corporation estimates that the permanent shutdown operation would cost $750,000. The estimated annual cost to maintain the plant in a safe shutdown condition is $50,000. This cost includes isolating the plant with suitable fencing and monitoring of the area by guards.

20.3 Source of Funds and Sharing of Costs

According to the Crystal River Unit 3 Participation Agreement executed by the co-owners, each co-owner is obligated to share in total operating costs of the unit (and in the shutdown and maintenance costs) in the same percentage as its percentage entitlement in the electrical capacity and output of the plant shown in Table 20.1.

The Florida Power Corporation's source of funds required for its ninety percent share of the plant's total operating expenses and for shutdown and maintenance expenses will be revenues derived from the sale of electric power to its retail and wholesale customers. As explained in Section 20.2 above, Florida Power Corporation's recent actual unit sales price is substantially higher than the estimated unit cost of operating the facility for the first five years of commercial operation. In addition, it is reasonable to assume that the Florida Power Corporation will receive future rate adjustments to compensate for increased costs of operating its system. The Florida Public Service Commission recently allowed Florida Power Corporation a $45.1 million permanent retail rate increase effective August 22, 1975, designed to increase total operating revenues by 12.5 percent.

As provided by the Participation Agreement, the additional ten percent of the total operating expenses and shutdown and maintenance expenses will be provided in the aggregate by the municipal and cooperative co-owners.

The source of funds for these expenses will be revenues from the sale of electric power to the customers of each system. The Florida Power Corporation will bill each additional co-owner monthly for its proportionate share of the total estimated

costs for the following month adjusted for differences between the previous month's estimate and actual charges.

The State of Florida's Joint Power Act of 1975 and its Revenue Bond Act of 1953 together empower the eleven additional co-owners to establish and periodically revise rates in order to fully recover all costs of operation including those related to Crystal River, Unit 3. Such rates are not subject to regulation by any authority other than the co-owners themselves. Under the terms of the Wholesale Power Contracts between Seminole Electric Cooperative, Inc. and its twelve member systems, Seminole is the exclusive power supplier to the systems and it also has the authority to establish rates to fully cover all costs of operation including those related to Crystal River, Unit 3.

20.4 Conclusion

Based on the preceding analysis, we have concluded that the Florida Power Corporation, the City of Alachua, the City of Bushnell, the City of Gainesville, the City of Kissimmee, the City of Leesburg, the City of New Smyrna Beach and Utilities Commission, City of New Smyrna Beach, the City of Ocala, the Orlando Utilities Commission and City of Orlando, the Sebring Utilities Commission, the Seminole Electric Cooperative, Inc., and the City of Tallahassee are financially qualified to operate the Crystal River Unit 3 Nuclear Generating Plant and, if necessary, to permanently shutdown the facility and maintain it in a safe shutdown condition.

22.0 CONCLUSIONS

Based on our evaluation of the application as set forth above, we have concluded that the issuance of an operating license, which limits plant operation to the refueling mode and the cold shutdown mode (Operational Modes 5 and 6 as defined in Table 1.1 of the Technical Specifications) will not be inimical to the common defense and security or to the health and safety of the public. We also reaffirm our conclusions as stated in our Safety Evaluation Report issued on July 5, 1974.

We conclude that with the repair of the containment dome, the construction of the facility has been substantially completed in accordance with the requirements of Section 50.57(a)(1) of 10 CFR Part 50.

Before the limitations on plant operation can be removed, the outstanding matters discussed herein, and itemized below, must be satisfactorily resolved. Further, the applicant will be required to satisfy the applicable provisions of 10 CFR Part 140.

The outstanding matters which must be satisfactorily resolved, and which will be addressed in a future supplement to our Safety Evaluation Report, are as follows:

(1) emergency core cooling analysis to meet the criteria described in Section 50.46 of Part 50 in accordance with Appendix K to 10 CFR Part 50 of the Commission's regulations;

(2) provisions for redundant safety grade low water level indication for the Borated Water Storage Tank;

(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;

(4) evaluation of the modifications to the containment spray system to assure that the spray pumps can operate within design limits in the containment spray mode of operation;

(5) evaluation of the final report regarding the repairs to the containment dome and the structural integrity test of the containment; and

(6) evaluation of the engineering hot channel factor for a replacement fuel bundle.

APPENDIX A

CHRONOLOGY

CONTINUATION OF CHRONOLOGY OF RADIOLOGICAL REVIEW

132.	December 4, 1974	Letter to applicant requesting commitment for their quality assurance program.
133.	January 13, 1975	Letter to applicant transmitting Supplement No. 1 to the Safety Evaluation Report.
134.	January 29, 1975	Letter to applicant advising that BAW-10095 report is acceptable.
135.	January 30, 1975	Letter to applicant advising that BAW-10035 report is acceptable.
136.	February 1, 1975	Amendment No. 44 filed.
137.	February 28, 1975	Letter from applicant concerning quality assurance program.
138.	February 28, 1975	Letter from applicant regarding WASH-1270, Anticipated Transients Without Scram, as related to Crystal River, Unit No. 3.
139.	February 12, 1975	Letter to applicant requesting description of program to prevent degradation of quality.
140.	February 12, 1975	Letter to applicant extending construction completion date and staff evaluation.
141.	March 3, 1975	Letter from applicant with information regarding quality assurance program.
142.	April 8, 1975	Letter from applicant for proposed Fuel Safety Analysis Report - revision.
143.	May 12, 1975	Letter to applicant transmitting schedule for implementation of B&W Standard Technical Specifications.
144.	May 13, 1975	Letter to applicant requesting additional information
145.	May 15, 1975	Meeting with applicant to implement B&W Standard Technical Specifications to the facility design.
146.	May 28, 1975	Letter from applicant advising information requested for secondary feedwater system will be provided by November 30, 1975.
147.	June 10, 1975	Letter to applicant requesting additional financial information.

148.	July 1, 1975	Meeting with applicant for implementation of applicant's interim project management team and implementation of Regulatory Guide 1.79.
149.	July 2, 1975	Meeting with applicant to implement B&W Standard Technical Specifications to the facility design.
150.	June 17, 1975	Letter from applicant advising of deficiency in performance of hydraulic reactor spray system.
151.	June 17, 1975	Letter to applicant requesting their conformance with Appendix I.
152.	June 19, 1975	Letter to applicant regarding meeting with staff to discuss Regulatory Guide 1.79.
153.	June 20, 1975	Letter from applicant advising proposed joint ownership for Crystal River, Unit 3 will take place in July 15, 1975.
154.	July 7, 1975	Letter from applicant furnishing information concerning organizational changes.
155.	July 10, 1975	Amendment No. 45 filed.
156.	July 10, 1975	Letter to applicant regarding deficiency in reactor building spray system.
157.	July 17, 1975	Letter from applicant changing execution date for joint ownership to July 31, 1975.
158.	July 24, 1975	Letter from applicant advising topical report will be incorporated in the Final Safety Analysis Report.
159.	August 11, 1975	Letter to applicant requesting additional information.
160.	August 11, 1975	Letter from applicant requesting license amendment for Plutonium-Beryllium neutron source.
161.	August 12, 1975	Letter to applicant requesting additional information for emergency core cooling system pressure calculations.
162.	August 15, 1975	Letter from applicant providing information on the Auxiliary Feedwater System in order to comply with single failure criteria.
163.	August 20, 1975	Meeting with applicant for review of the initial draft of the facility Standard Technical Specifications.
164.	August 21, 1975	Letter from applicant updating information on joint ownership for Crystal River, Unit 3.
165.	August 27, 1975	Amendment No. 46 filed.
166.	August 26, 1975	Letter from applicant concerning missile shields to cover spent fuel pools.
167.	September 3, 1975	Letter to applicant referencing summary of meeting for initial review of Standard Technical Specifications.

168.	September 3, 1975	Letter from applicant providing information on the effects of secondary systems fluid flow instability.
169.	September 16, 1975	Letter to applicant advising they provide schedule for meeting Regulatory Guide 1.79.
170.	September 19, 1975	Letter from applicant furnishing additional information on the emergency core cooling system.
171.	September 22, 1975	Letter from applicant regarding testing for conformance with Regulatory Guide 1.79.
172.	October 9, 1975	Letter to applicant advising response to staff regarding sump test for vortex control.
173.	October 13, 1975	Letter from applicant providing design information on substituted valves.
174.	October 17, 1975	Letter from applicant regarding schedule for pre-operational testing of emergency core cooling system.
175.	October 17, 1975	Letter from applicant regarding qualification testing for balance of plant Class IE equipment.
176.	October 28, 1975	Amendment No. 47 filed.
177.	November 5, 1975	Letter to applicant advising the interim installation of substitute valves acceptable.
178.	November 10, 1975	Letter from applicant providing technical qualifications of operations personnel.
179.	November 11, 1975	Letter to applicant transmitting report evaluation.
180.	November 13, 1975	Letter to applicant transmitting Amendment No. 1 to provisional Construction Permit CPPR-51.
181.	November 14, 1975	Letter to applicant advising of potential safety concern regarding design of reactor pre-sure vessel support systems for pressurized water reactors.
182.	November 21, 1975	Letter from applicant submitting interim six-month meteorological report (January 1, 1975 through June 30, 1975).
183.	November 26, 1975	Letter from applicant providing technical qualifications of station operating personnel.
184.	November 26, o975	Letter from applicant transmitting enlarged electrical drawing.
185.	December 5, 1975	Letter from applicant regarding failure of fuel assembly 3A33.
186.	December 12, 1975	Letter from applicant regarding Standard Technical Specifications.
187.	December 8, 1975	Letter to applicant requesting additional information for emergency core cooling system.

188.	December 18, 1975	Letter from applicant providing additional information for emergency core cooling system sump test.
189.	December 22, 1975	Letter from applicant providing information on reactor vessel support loadings.
190.	February 2, 1976	Letter from applicant providing response to qualification testing for protection system instrumentation.
191.	February 5, 1976	Letter from applicant providing proposed revision to Final Safety Analysis Report including radiological emergency plan.
192.	February 11, 1976	Letter from applicant providing information on line losses during recirculation mode in reactor building sump.
193.	February 18, 1976	Letter from applicant submitting Revision No. 2.
194.	February 18, 1976	Meeting with applicant to discuss the facility emergency and industrial security plans.
195.	February 19, 1976	Letter to applicant providing guidance for meeting requirements of Appendix I.
196.	March 2, 1976	Letter to applicant for requests for additional information.
197.	March 16, 1976	Letter from applicant providing information on borated water storage drawdown tests.
198.	March 18, 1976	Meeting with applicant to discuss the facility industrial security plan.
199.	March 19, 1976	Letter from applicant providing additional information on Revision No. 2 to the security plan.
200.	March 19, 1976	Letter from applicant furnishing information on hot functional testing.
201.	March 24, 1976	Letter to applicant advising staff will be available in Atlanta, Georgia on April 6, 1976 to discuss Appendix I.
202.	March 29, 1976	Letter from applicant providing information concerning Appendix I criteria.
203.	March 30, 1976	Amendment No. 48 filed.
204.	March 29, 1976	Letter from applicant providing report on fuel assemblies.
205.	April 6, 1976	Letter to applicant advising condition of operating license for fully implementing all provisions of staff approved industrial security plan.
206.	April 18, 1976	Meeting with applicant to discuss the borated water storage drawdown test.
207.	April 21, 1976	Letter from applicant notifying staff of concrete separation on the containment dome.

208.	May 7, 1976	Letter from applicant providing analysis of clad-creep-collapse.
209.	May 7, 1976	Letter to applicant transmitting draft mode technical specifications for pressurized water reactors.
210.	May 10, 1976	Letter to applicant advising that adopted amendments to Parts 2, 50, and 51 for required changes pertaining to applications and amendments for operating licenses.
211.	May 12, 1976	Meeting with applicant for status report on concrete separation in the reactor building dome.
212.	May 13, 1976	Letter to applicant regarding separation of concrete on the containment dome.
213.	May 24, 1976	Letter to applicant regarding approved first cycle operation of fuel assembly 3A33.
214.	May 28, 1976	Letter from applicant providing Appendix I analysis for Crystal River, Unit 3.
215.	June 11, 1976	Letter from applicant with enclosed report on containment dome delaminations.
216.	June 18, 1976	Meeting with applicant to discuss concrete separation in the reactor building dome.
217.	June 22, 1976	Letter to applicant advising that requested waiver for operator examinations will be granted.
218.	June 24, 1976	Letter from applicant providing Revision No. 2 to their Appendix I analysis.
219.	June 24, 1976	Letter from applicant providing information on reactor vessel support design.
220.	June 24, 1976	Letter from applicant regarding massive revision and imposition of Standard Technical Specifications.
221.	July 12, 1976	Letter from applicant providing information on sodium hydroxide injection for borated water storage tank drawdown analysis.
222.	July 28, 1976	Letter from applicant providing information on quality assurance program.
223.	August 10, 1976	Letter from applicant providing Supplement No. 1 to reactor building dome delamination report.
224.	August 16, 1976	Letter from applicant providing revised and amended pages to reactor building dome delamination report.
225.	August 23, 1976	Letter to applicant regarding the status of the B&W redesign of surveillance specimen holder tubes for B&W 177 fuel assembly plants.
226.	August 25, 1976	Letter to applicant enclosing Federal Register Notice (41FR31521).

227.	September 9, 1976	Letter from applicant providing electrical drawings for emergency core cooling systems.
228.	September 15, 1976	Letter from applicant providing information on the reactor building dome delamination.
229.	September 15, 1976	Letter from applicant providing evaluation in the initiation of repairs to reactor building dome.
230.	September 22, 1976	Letter from applicant transmitting amended pages for reactor building dome delamination report.
231.	September 24, 1976	Letter from applicant providing results of their review and comparison of operational quality assurance program.
232.	September 29, 1976	Letter from applicant providing information on integrated irradiation program for reactor vessel surveillance specimens.
233.	October 1, 1976	Letter to applicant regarding equipment failures during degraded grid voltage condition and request for additional information.
234.	October 1, 1976	Letter to applicant on verification for compliance with Appendix G pressure-temperature limits during startup and shutdown.
235.	October 1, 1976	Letter to applicant requesting additional information on reactor building dome delamination report.
236.	October 5, 1976	Letter from applicant transmitting structural integrity test results.
237.	October 6, 1976	Amendment No. 49 filed.
238.	October 8, 1976	Letter from applicant transmitting acceptance criteria used in evaluation of structural integrity test results.
239.	October 11, 1976	Letter from applicant providing reactor building sump test head losses.
240.	October 15, 1976	Letter from applicant providing information on hydraulic snubber technical specifications.
241.	October 26, 1976	Letter from applicant regarding reevaluation of the protection program.
242.	October 28, 1976	Letter from applicant providing corrected Supplement No. 2 report on reactor building dome delamination report.
243.	October 29, 1976	Letter to applicant providing information concerning reactor overpressurization.
244.	October 28, 1976	Letter from applicant providing predicted values for strain gauge readings obtained in cap of dome.

245.	October 29, 1976	Letter from applicant providing proposed change to Amendment No. 50.
246.	October 29, 1976	Letter from applicant furnishing information regarding equipment failures during a degraded grid voltage condition.
247.	November 5, 1976	Letter from applicant transmitting preliminary test report for structural integrity test.
248.	November 15, 1976	Letter from applicant advising that original feedwater valves (FWV-161 and FWV-162) have been received, installed, and tested.