June 6, 1968

SAFETY EVALUATION

BY THE

DIVISION OF REACTOR LICENSING

UNITED STATES

ATOMIC ENERGY COMMISSION

IN THE MATTER OF

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

TABLE OF CONTENTS

APPENDICES

1.0 INTRODUCTION

The Florida Power Corporation (applicant) has filed an application dated August 10, 1967, with subsequent amendments, requesting a license to construct and operate a pressurized water reactor, identified as Crystal River Unit 3 Nuclear Generating Plant, in Citrus County, Florida.

The proposed reactor is designed to operate initially at core power levels up to 2452 Mw thermal. The applicant anticipates, however, that the reactor will ultimately be capable of operating at a core power level of 2544 Mw thermal. The design of the major systems and components of the proposed facility, including the emergency core cooling systems and the containment structure, which bear significantly on the acceptability of the facility under the site criteria guidelines identified in 10 CFR Part 100 of the Commission's regulations, have been analyzed and evaluated by the applicant and the regulatory staff at the higher power level of 2544 Mw thermal. The thermal and hydraulic characteristics of the reactor core were analyzed and evaluated at the initial power level of 2452 Mw thermal. Before operation at any power level above 2452 Mwt is authorized, the regulatory staff must perform a safety evaluation to assure that the core can be operated safely at the higher power level.

A technical safety review of the proposed plant has been performed by the Commission's regulatory staff, based on the applicant's Preliminary Safety Analysis Report (PSAR) and five subsequent amendments all of which are contained in the application. This technical safety review or evaluation of the preliminary design of the proposed plant was accomplished by the Division of

Reactor Licensing and various AEC consultants, as requested. Within the Division of Reactor Licensing, the Reactor Projects group was assigned primary responsibility for the review. Assisting this group in its review were personnel within the Division representing various special technical disciplines from the Reactor Technology and Reactor Operations groups. Their work was coordinated by the Reactor Projects group.

In the course of our review of the application, we held meetings with representatives of the applicant, the nuclear steam system supplier (the Babcock & Wilcox Company) and the architect-engineer (Gilbert Associates) to discuss the proposed plant and to clarify the technical material submitted. As a consequence of these meetings, additional information was submitted as various amendments to the application. A chronology of the meetings and principal correspondence is given in Appendix B to this evaluation. Reports by our consultants on meteorology, hydrology and geology, seismicity, seismic design, hurricane effects, and environmental considerations are included in Appendices C through H to this evaluation.

In addition, the Commission's Advisory Committee on Reactor Safeguards (ACRS) has also considered this project and has met and discussed it with both the applicant and the regulatory staff. The report of the ACRS is included as Appendix A.

The review and evaluation of the proposed design and construction plans of the applicant at this, the construction permit stage, is only the first stage of a continuing review of the design, construction and operation of the proposed nuclear power plant. Prior to issuance of an operating license for the facility, we will review the final design thoroughly to determine that all the Commission's safety requirements have been met. The unit would then be operated only in accordance with the terms of the operating license and the Commission's regulations and under the continued scrutiny of the Commission's regulatory staff.

The issues to be considered, and on which findings must be made by an atomic safety and licensing board before the requested license may be issued, are set forth in the Notice of Hearing issued by the Commission and published in the Federal Register on June 1, 1968 (33 FR 8235).

2.0 SITE

2.1 Description

The Crystal River Unit 3 will be located on the Gulf of Mexico about 70 miles north of Tampa, Florida and 7-1/2 miles northwest of the town of Crystal River, Florida.

The nearest population center of 25,000 or more is Gainesville, Florida, 55 miles NNE from the site. There are no residents within a 3-1/2 mile radius. No inhabitants are expected within 3 miles for the projection to year 2015. The 1967 population distribution figures for the Crystal River site include 3300 people within a 10-mile radius and slightly over 6000 within a 20-mile radius. A significant increase in the 5-10 mile zone population density is projected for the 40-year life of the plant, primarily as a result of an increase in the population of the town of Crystal River from slightly over 3000 people to over 25,000 people.

The site is characterized by a 4400-foot exclusion radius and encompasses 4738 acres, all of which are wholly owned by the applicant. The low population distance, computed from the present population center of Gainesville, is 41 miles. However, we have considered the low population distance to be 5 miles to take into account the potential future growth of the town of Crystal River.

Cooling water for the reactor is obtained from the nearby Gulf of Mexico. Sea-water intake and discharge canals, now at the site for two fossil-fired units, will be extended to the nuclear Unit 3. Condenser water and water for auxiliary and emergency cooling requirements is obtained from and discharged to these canals.

2.2 Meteorology

The applicant has assumed conservative diffusion parameters in assessing the consequences of releases of airborne radioactive materials. The Environmental Meteorology Branch of the Institute for Atmospheric Sciences has reviewed the proposed meteorological assumptions and indicates in its report (Appendix C, attached) that the model is conservative. We also conclude that the model is conservative. A meteorological data collection program was initiated at the site in 1967, and will be continued for at least 2 years during the preoperational period. The program is acceptable for confirmation of the meteorological assumptions.

2.3 Geology and Hydrology

The site geology relevant to the plant foundations is characterized by limestone which has been subjected to solutioning in the past. Studies have been performed to determine methods for groundwater control and for means of filling solution voids of significant extent and secondarily providing densification of zones of loose materials. The applicant proposes a consolidation grouting program to fill the solution channels, confine potential settlement-inducing zones, and minimize solution rates. In addition, a curtain grout around the foundation area will control groundwater. The applicant developed a grouting procedure during the construction of nearby Unit No. 2 (fossil-unit) which he proposes also to use for the nuclear plant. This procedure uses what is called a split-spaced stage grouting technique.

The concept involves drilling primary holes, making initial grout injections, then grouting again in split-spaced holes (drilled midway between original holes). This procedure is repeated until the cubic feet of grout injection per lineal foot of injection hole (grout take) is decreased to a predetermined value. Stage grouting refers to an intermittent grout injection where succeeding injections form on the base laid by previous injections.

Our consultants, the U.S. Geological Survey and A. J. Hendron of Nathan M. Newmark Consulting Engineering Services, have reviewed the PSAR material and have visited the site. The U.S. Geological Survey report, contained in Appendix D of this Safety Evaluation, concludes, and we agree, that the grouting program should preclude existence of any large cavities in the limestone that underlie the plant structure, and that the resulting grout-stabilized rock should provide an adequate foundation for the proposed nuclear facility. The report of Newmark Consulting Engineering Services, included in this Safety Evaluation as Appendix F, concludes, and we agree, that the modified split-spaced hole procedure utilized on fossil Unit No. 2 will be adequate for the foundation of nuclear Unit No. 3.

There are no unusual hydrologic problems with this site other than those associated with hurricane effects, which are discussed in Section 2.5 of this Safety Evaluation. Our hydrologic consultant, also the U.S. Geological Service, concludes, and we agree, that the reactor at this location is not likely to affect the fresh water resources of the area.

2.4 Seismology

The applicant has studied the seismicity of the site. He has concluded that a value of 0.05 g horizontal acceleration for the "design earthquake" and a corresponding value of 0.10 g for the "maximum probable earthquake" should be used in the design. We consider these values to be acceptable. The U.S. Coast & Geodetic Survey has reviewed the PSAR material and recommends the same acceleration values. The report is attached to the safety evaluation as Appendix E.

2.5 Hurricane Effects

The site proximity to the Gulf of Mexico required that studies on hurricane effects and wave action be performed. The applicant calculated the hurricane effects for a "Maximum Probable Hurricane." Wave periods and heights and hurricane tide levels were calculated. Model studies using these analyses with a model of the proposed plant design were performed.

The analysis showed that a plant elevation of 30.4 feet above mean low water level (MLW) would give full protection against the postulated hurricane tides and wave action for all components which must operate for a safe and orderly shutdown. The model studies showed that an additional 1 to 1-1/2 feet of water rise could be tolerated before wave runup could reach the site grade elevation.

In the analysis of the maximum probable hurricane the still water level computed was 21.4 feet above mean low water level, and the wave height was an additional 9.0 feet. The proposed plant elevation is 30.4 feet above mean low water level. Immediate unperturbed ground elevation is about 10.0 above mean low water level.

Our consultant, the Coastal Engineering Research Center (CERC) of the Army Corps of Engineers has reviewed the proposed design and concludes that the applicant has used design water levels, wave characteristics and wave runup criteria comparable with MPH occurrence as now defined. The CERC report is attached to this Safety Evaluation as Appendix G.

Additional studies are in progress by the Hydrometeorological Branch of the Environmental Sciences Services Administration (ESSA) on sizing the Maximum Probable Hurricane (MPH) for various Atlantic and Gulf coastal sites. We intend to evaluate the proposed plant design relative to hurricane effects as soon as the revised ESSA report is available. The applicant has stated, and we will require, that the plant protection will conform to the applicable portions of revised ESSA criteria.

2.6 Environmental Radioactivity Monitoring

A preoperational environmental radioactivity monitoring program has been proposed by the applicant. The sampling program will be initiated at least 2 years prior to Unit No. 3 startup and will include samples of Gulf and well water, soil, air particulate, animal thyroids, fish, shellfish, and bottom sediments. Post-operational monitoring will be similar to the preoperational program. Reconcentration of specific radionuclides by the local aquatic biota has been considered. The applicant recognizes that the reconcentration factor must be considered in conjunction with Part 20 limits in assuring that discharges are within acceptable limits.

A copy of the application was forwarded to the Fish & Wildlife Service for their review. Comments of the Fish & Wildlife Service are attached as Appendices H-1, H-2, H-3 (original report and two supplements).

We conclude that the environmental program as proposed is acceptable.

3.0 NUCLEAR STEAM SYSTEM DESIGN

3.1 Summary Description

The nuclear steam supply system consists of a light water moderated and cooled pressurized water reactor (PWR) which transfers reactor heat to two once-through steam generators from which steam passes to a turbine-generator unit. The low-enrichment UO_2 pellet fuel is held in zirconium rods 0.4 inch in diameter and about 12 feet in length. The fuel rods are held in place by perforated-can fuel assemblies which have eight lateral grid spacers over the 12-foot length in addition to the two end fittings. Each assembly contains 208 fuel pins, 16 control pin guide tubes and one in-core instrument guide tube.

The core, comprised of 177 of these fuel assemblies, rests on the lower grid plate which is attached to the core support barrel in turn attached to the reactor vessel wall near the top of the vessel. The core obtains lateral support from the center grid plate, located at the top of the assemblies. An upper grid plate, above the core, provides lateral guidance for the control rod assemblies.

Reactivity control is accomplished by 69 control rod assemblies and by a liquid poison (boric acid) in the reactor coolant. Each control rod assembly consists of 16 stainless-steel tubes, containing a silver-indium-cadmium neutron-absorbing alloy, which are connected to a "spider" assembly at the top so that the 16 absorber-filled tubes act as a unit. The control rod assembly is withdrawn and inserted by a rack and pinion drive assembly mounted on the reactor vessel head and driven through a magnetic clutch by a synchronous motor. If a rapid reactor shutdown is desired, the control rod assembly may be dropped by gravity into the core by releasing the magnetic clutch. As the fuel is depleted, criticality is maintained by removing the liquid poison from the system by a chemical addition and control system.

The nuclear flux level is monitored by neutron detectors external to the reactor vessel and by 52 in-core detectors. Either the nuclear flux level, high or low reactor system pressure, high coolant temperature, or low coolant flow rate can initiate a reactor trip through the reactor protection instrumentation. The trip de-energizes the magnetic clutches on the control rods and scrams the reactor.

Water is heated (from about 555° F to about 600° F at 2200 psi) while passing upward through the reactor core and exits from the reactor vessel through two 36-inch diameter lines near the top of the vessel. Each "hot leg" enters the top of a once-through steam generator. The primary coolant passes downward through the steam generator within a bank of tubes where it is cooled by water and steam (at about 570° F and 910 psig) on the shell side. The coolant is returned to the reactor vessel from the bottom of the steam generators through four "cold legs" (two from each steam generator). Each cold leg contains a reactor coolant pump which provides the circulatory driving force.

Steam generated on the shell side of the steam generators is superheated by about 35° F before passing through steam lines to a turbine-generator unit outside the containment building. After passing through the turbine, the low-pressure steam is condensed in the turbine condenser and returned as feedwater to the steam generators by electrically driven condensate pumps and condensate booster pumps in series with steam-driven feedwater pumps.

The pressure vessel and primary system piping, steam generators, control rod drives, instrumentation, core internals and the first core fuel will be supplied by the Babcock & Wilcox Company (B&W). The steam turbine will be purchased from the Westinghouse Electric Corporation.

The B&W nuclear steam system design is essentially identical to those approved for use by the Duke Power Company at its Oconee Nuclear Station and for the Metropolitan Edison's Three Mile Island Station.

The only subsystem of the nuclear steam system which differs substantially in design concept from current pressurized water reactor practice and experience is the once-through steam generator which provides slightly superheated steam to the turbine-generator. (The once-through steam generator was also approved for use in the Oconee and Three Mile Island Nuclear Stations.) Other subsystems such as the rack and pinion control rod drives and the instrumentation are new designs but are based on experience with similar concepts. These systems will be discussed in more detail in following sections of this report.

3.2 Nuclear Design

The light water moderated and cooled core has been designed to allow operation at 2452 Mw thermal to a maximum fuel burnup of 55,000 megawatt days per metric ton of uranium. The total clean cold excess reactivity is about 30% delta k/k. About 10% delta k/k is controllable by the control cluster assemblies and the remainder by soluble poison. The reactor can be made subcritical by 1% delta k/k with the highest-worth control assembly stuck out of the core, at hot conditions, by inserting the other 68 control assemblies. A similar margin can be obtained at cold conditions by insertion of soluble poison. The reactivity worth of the control rod assemblies and the rate at which reactivity can be added by the assemblies or by the soluble poison system is limited to prevent damage to the primary system or the fuel. The nuclear design objectives and limits are similar to other pressurized water reactors now under construction.

The core is predicted to have a positive moderator temperature coefficient of reactivity during part of the first fuel cycle. The positive coefficient has been calculated by the applicant to be about 1.0×10^{-4} delta $k/k/^{o}F$ at the beginning of core life. This is calculated to correspond to a maximum 0.5% delta k/k in reactivity which could be inserted by a reduction in moderator density. If this reactivity were inserted during a loss-of-coolant accident caused by the break of a large system pipe, about 2.1 full power seconds of energy would be released. The resulting peak fuel temperature caused by such a transient would be less than 2000^{o} F, based on the present calculations. An acceptable value of the positive moderator temperature coefficient will be set at the operating license stage, based on the final design and more refined calculations. The applicant has agreed to reduce this coefficient by the addition of stainless-steel shims if necessary. Although we are continuing to evaluate the magnitude of the energy added during a loss-of-coolant accident, we believe that the proposed core design criteria are acceptable at this time. The applicant has agreed, if necessary, to eliminate or reduce the positive coefficient to bring the consequences of the applicable accident within acceptable limits.

The applicant's calculations indicate the stability margin with respect to xenon oscillations is least for the axial direction and that azimuthal and radial oscillations are not expected. Further analysis will be made using final values of core properties and if it is found that oscillations could occur, a method for controlling the oscillations will be employed. Calculations have been made to illustrate the ability of control rod assemblies with a short poison section to control a divergent xenon oscillation. Since xenon oscillations involve relatively slow flux changes, and since the flux imbalance could be detected by the proposed instrumentation, we believe that this method of control is feasible and that analytical and, if necessary, control techniques can be developed prior to the operating stage. Manipulation of the normal control rod assemblies or power reduction can also be used to prevent or correct, to some extent, the undesirable effects of xenon oscillations.

3.3 Mechanical Design of Reactor Internals

The reactor internals will be designed to withstand steady-state and anticipated operational transient loads. In addition, the internals will be designed to resist the combined effects of seismic disturbances and loss-of-coolant accident forces resulting from a primary system pipe break. The applicant has performed the analysis of combined seismic and blowdown forces based on preliminary estimates of pressure differential time histories. The final loadings will be submitted to us for review when available later this year. We and our seismic design consultant, Nathan M. Newmark, have reviewed the proposed loading combinations and deformation limits associated with combined seismic and blowdown forces. The reactor coolant system and all other Class I (seismic) mechanical systems will be designed to withstand normal design loads of mechanical, hydraulic and thermal origin plus design earthquake loads within normal code allowable stresses. In addition, Class I systems and components will be designed to withstand, with no loss of safety function capability, the concurrent accident and maximum probable earthquake loads. Primary membrane stress intensities under the most severe load combinations will not exceed 2/3 of the stresses corresponding to the uniform strain value at operating temperature. This criterion results in the allowable strains less than 20% of uniform strain for all pertinent materials in the unirradiated condition. We and our seismic design consultant consider the proposed stress criterion proposed to provide an adequate margin of safety.

The fuel assemblies are designed for steady-state and transient conditions under the combined effects of flow-induced vibration, reactor pressure, fission gas pressure, fuel growth and thermal strain. The cold-worked Zircaloy-4 cladding is designed to be free-standing. The fuel rod spacers are designed to maintain spacing between the fuel rods but to permit thermal expansion of the rod. Structural stability is obtained from a perforated can assembly around the 15 by 15 array (which includes 16 Zircaloy control pin guide tubes and one in-core instrument guide tube as well as the Zircaloy-clad UO_2 pellet fuel).

The control rod assembly travel is designed so that the control pins are always engaged in the fuel assembly control pin guide tubes, ensuring that the control assembly can be dropped into the core when required. Each pin of the cluster is also guided above the core by tubes slotted to allow passage of the spider connection. The internals are designed to ensure that the dynamic loading resulting from a loss-of-coolant blowdown will not prevent insertion of the control rod assemblies. The stresses imposed on the control rod during scram are minimized by a snubbing mechanism in the rod drive housing and by designing the assembly for the deceleration loads.

The design criteria for the mechanical design of internals, fuel assemblies and control assemblies are adequate.

3.4 Thermal and Hydraulic Design

The reactor core is designed to operate at a steady-state power level of 2452 megawatts thermal corresponding to an average linear heat generation rate of 5.4 kw per foot of fuel rod and a peak of 17.5 kw per foot. The calculated maximum fuel temperature is about 4160° F and the average fuel temperature about 1385° F.

Although the turbine-generator unit and other equipment are sized for a higher core power level (2544 Mwt) and the fission product release studies are based on this higher power level, the application is for a core power level of 2452 Mwt and we have reviewed the thermal-hydraulic characteristics of the core at this power level.

The reactor core is designed (1) to prevent fuel melting at the design overpower of 114% (2795 Mwt), (2) to provide a high degree of assurance that no departure from nucleate boiling (DNB) will be experienced in the core, and (3) to maintain steam voids in the hottest channel at a level well below the threshold of flow instability. The design overpower is the highest reactor power which could result from foreseeable reactor operational transients which are terminated by reactor protective systems action (action is initiated at 107.5% full power).

The thermal and hydraulic design evaluation presented in the PSAR made use of the BAW-168 heat transfer relationship to establish that DNB would not be reached at the 114% overpower condition. A probability study was included in the analysis as a means of demonstrating the sensitivity of the analysis to the various input parameters and to allow an expression of the fraction of the core endangered when at various hot channel DNB ratios. B&W substantiated the design by the results of rod bundle burnout tests of similar geometry but with axially uniform heating. These results were corrected to fit the actual nonuniform case by use of a correction factor obtained from single-rod burnout data. The applicant also performed calculations using the Westinghouse W-3 correlation to confirm that the thermal design limits are met.

Axially nonuniform bundle tests, similar in geometry to the proposed design, are being run as part of the research and development program at B&W and the results of these tests will be applied to the final thermal design. We believe that the allowable design heat flux should be designated as a research and development item if the design is to be based on the B&W heat transfer data. On the basis of the preliminary research results submitted it appears that B&W will be able to justify the chosen physical parameters and design limits on the basis of its program of rod bundle burnout tests. In any event the design is acceptable on the basis of the W-3 correlation.

Since the B&W design has four inlet loops and only two outlet pipes, the coolant distribution within the reactor vessel must therefore be investigated and the associated pressure drops established. The applicant has stated that a research and development program is underway to measure flow distribution in the core, fluid mixing in the vessel and core, and the distribution of pressure drop within the vessel. These tests will be conducted on a 1/6 scale model of the vessel and internals. In addition, flow distribution, pressure drop, and mixing data will be obtained with a full scale fuel bundle test assembly and on various models of reactor flow cells.

We have reviewed the development program as described above and believe that there is reasonable assurance that the scale-model testing and the full-scale fuel bundle testing will provide the information necessary for approval of the design at the operating license review stage.

3.5 Control Rod Drive Design

The drive mechanism proposed is a rack and pinion device driven by a synchronous stepping motor through a worm gear reducer, unidirectional clutch, magnetic clutch, drive shaft, and miter gear set. The drive is operated in primary coolant up to the magnetic clutch where a buffer seal and rotary seal prevent leakage of primary coolant.

The drive motor assembly utilizes a worm gear reducer to prevent torque from being transferred to the drive motor in the event an upward force is applied to the rack. A unidirectional clutch will be provided within the magnetic clutch to prevent upward movement of the rack without a rod withdrawal signal from the control system.

Normal rod withdrawal and insertion requires that the magnetic clutch be energized. Scram is accomplished by deenergizing the clutch. The entire drive train from the control assembly, the rack and pinion, and the clutch all move during a scram.

The components of the drive that operate in reactor coolant will be capable of performing their function at 650° F. The seal water injection to the buffer seal is expected to maintain the drive components at a lower temperature. The applicant has proposed a development program to fully test the proposed design to demonstrate that the design objectives are met.

Our review of the proposed design indicates that no unusual problems are apparent. The applicant's design objectives and the development program should provide an acceptable control rod drive mechanism prior to the operating license review.

3.6 Instrumentation and Control

3.6.1 Reactor Protection System

The reactor protection system monitors vital process variables and automatically causes reactor shutdown when predetermined conditions established for each variable have been exceeded. The variables monitored include (1) high reactor power, as measured by neutron flux, (2) low reactor coolant flow, (3) high reactor outlet temperature, and (4) high or low reactor pressure.

The protection system consists of four identical and independent protection channels, each terminating in a bistable and trip relay. Each of the above variables is monitored by four channels which are coincident and redundant.

The nuclear instrumentation has eight channels of neutron information divided into three ranges of sensitivity: source range, intermediate range, and power range. The three ranges combine to give a continuous measurement of reactor power from source level to approximately 125% of full power, or ten decades of information. A minimum of one decade of overlapping information is provided.

The source range instrumentation channels consist of two redundant count rate channels, each using proportional counters as sensors. These channels are not associated with a protection function; however, they do provide an interlock function (a control assembly withdrawal hold and alarm on high startup rate).

The intermediate range instrumentation has two log-N channels, each using identical gamma-compensated ion chambers as sensors. Reactor trip initiation is provided by these channels.

The power range instrumentation consists of four linear level channels using three uncompensated ion chambers per channel. The gain of each channel is adjustable, providing a means for calibrating the output against a reactor heat balance. Protective action consists of reactor trip initiation at preset flux levels.

Primary loop flow information is measured as a function of pressure drop by four independent sensors in each of the two hot legs. The outputs of the eight sensors are combined as pairs such that four independent total flow signals are derived. Each total-flow signal is fed to one of the four power range channels, thus creating four independent power/flow channels.

The logic of the power/flow channels is two-out-of-four, and the channels are independently connected to the reactor protection system logic channels in the same manner as the power range channels. The power/flow channels will initiate a reactor trip if the reactor power exceeds 107.5% full power or if a mismatch exists between power and coolant flow. In addition, each pump motor breaker has four contacts which are respectively connected to the reactor protection system such that a reactor trip occurs if less than three pumps are in operation.

There is one set of four pressure sensors and one set of four temperature sensors which respectively trip the reactor on high and low primary system pressure, and high coolant outlet temperature. The logic is two-out-of-four, and the instrument channels are independently connected to the four logic channels in the same manner as the power range channels. One pressure channel also provides a signal to the pressurizer pressure controller. The other three channels will provide trip action on a redundant basis should a failure disable the one common channel and simultaneously initiate a pressure transient. The ACRS has stated (Section 11.0) that the control and protection instrumentation should be separated to the fullest extent practicable. We will review the detailed final design, and implement this recommendation, at the operating license stage.

The nuclear and process instrument channels, by virtue of being redundant, can withstand any single failure without loss of protective function. The coincident logic permits testing during reactor operation. In addition, all instrument channels initiate a trip signal in the event of AC voltage loss. The final trip circuit of the proposed reactor protection system consists of a split DC bus fed from two station batteries.

The in-core instrumentation system, consisting of 52 in-core detectors located in the fuel assembly guide tubes, provides no automatic control or protection function. The system is located entirely within containment, thereby precluding the need for isolation of penetrations associated with the system. If xenon oscillations prove to be a problem in the final core design, part-length rods may be required. The in-core instrumentation system could then be used to supplement out-of-core information on xenon-induced core flux tilting to allow the operator to take proper corrective action. In the event that the plant computer which provides the in-core system readout is not operable, an alternate readout system is available for selected in-core detectors for xenon oscillation observation. The self-powered in-core neutron detector units are currently under test in the Big Rock Point Nuclear Power Plant.

We conclude that the reactor protection system is acceptable.

3.6.2 Engineered Safety Features Control and Instrumentation

The engineered safety features are automatically initiated as follows: (1) operation of the core emergency injection systems upon detection of low reactor coolant pressure or high reactor building pressure, (2) operation of the reactor building cooling and iodine removal systems upon detection of high reactor building pressure, (3) containment isolation upon detection of high reactor building pressure, and (4) closure of isolation valves on the purge lines which are directly open to the reactor building, on receipt of a high radiation signal or high reactor building pressure.

The engineered safety features' instrument channels do not control the parameters which they measure; i.e., there is separation of control and safety. Manual actuation capability, independent of the instrument channels, is provided.

The containment emergency cooler fans and motors are the only components which must operate in the containment atmosphere for an extended period of time after a design basis loss-of-coolant accident. The fan motors will be designed so that windings and bearing surfaces are protected against the accident environment. Motor housings will be designed to withstand 60 psi, and are cooled by an air-water heat exchanger connected to the nuclear services cooling water system. The winding insulation will have been demonstrated to withstand an accumulated radiation exposure greater than the expected lifetime and design basis accident exposures. Bearings will be of a seal type which will withstand the design basis accident pressure pulse and will also be cooled by the nuclear services cooling system. The applicant will review the bearing design after integration into the cooler system and determine the necessity for environmental tests. We will also assure ourselves that the fan motor housings have been conservatively designed and, depending on the final design, may require a prototype environmental test of the complete motor unit.

3.6.3 Reactivity Control

Reactivity control is maintained by movable control assemblies and by soluble poison (boric acid) dissolved in the reactor coolant.

The control rod drives will be designed so that (1) no single failure can cause an uncontrolled withdrawal of any rod, (2) no more than two control groups can be withdrawn at one time, (3) the withdrawal speed will be limited so as not to exceed 25 percent overspeed in the event of speed control faults, and (4) continuous position indication will be provided. Based on our analysis, we believe that the applicant's rod drive system criteria are acceptable, that no single failure in the control instrumentation can produce an excursion which will cause fuel damage and that the proposed rod drives can be built in accordance with these criteria.

Reactivity is also controlled by a permissive system which allows manual dilution of the primary system coolant boron concentration when a particular control rod group reaches the fully withdrawn position. Dilution is automatically terminated when the rod group, driven down by the servo, reaches a prescribed position, or when the integrated dilution flow has reached a preset maximum. These circuits will be designed in accordance with protection standards and no single failure will prevent automatic termination of dilution.

In summary, we conclude that the applicant's design criteria relating to reactivity controls are satisfactory and that the proposed preliminary designs conform to these criteria, and are therefore acceptable.

3.6.4 Radiation Monitoring System

The radiation monitoring system for this plant consits of three subsystems: area gamma monitoring, atmospheric monitoring, and liquid monitoring. The area monitors consist of those instrument channels which indicate general levels of radioactivity at selected locations in the plant. The atmospheric and liquid monitors consist of those instrument channels which measure radioactivity levels within specific plant processes and automatically initiate corrective action or indicate that corrective action should be taken.

The detectors selected for each location have sufficient ranges and sensitivities to provide readings within range during a design basis accident and will be located in close proximity to the points of releases or areas of most probable equipment failure. All instruments will receive power from the vital instrument buses thereby assuring their availability to perform their required function under accident conditions.

High radiation signals will be used to automatically shut off discharges from the liquid and gaseous waste disposal systems. The appropriate gas activity signals will also shut down the auxiliary, fuel handling and reactor building ventilation systems. The cooling water systems which remove heat from potentially radioactive sources will be monitored to detect accidental releases. The systems are the intermediate cooling water, nuclear services closed cooling water, spent fuel cooling water, plant liquid effluent line, and liquid waste discharge prior to dilution. The radiation monitoring systems proposed for the Crystal River Nuclear Station are acceptable. The ACRS has recommended (Section 11.0) that consideration should be given to development and utilization of instrumentation for prompt detection of a gross fuel element failure. Scoping studies on such a system have been initiated by B&W. We will review the results of that study when available.

3.7 Reactor Coolant System

3.7.1 Primary System

The reactor coolant is transferred to the top of the two once-through steam generators through two 36-inch lines from the upper reactor vessel plenum. Water is returned from the bottom of the steam generators to the vessel via four 28-inch lines. Circulation is provided by a single-speed, shaft-sealed pump in each of the four cold legs.

The applicant has stated that access for inspection can be gained to all internal surfaces of the primary vessel by removing vessel internals and that it will be possible to gain access to the external vessel surfaces although this would require the removal of thermal insulation. The scope and frequency of the inspection program will be reviewed at the operating license stage.

The applicant presented the results of an analysis of the thermal transient experienced by the hot reactor vessel wall when deluged with cold safety injection water after a loss-of-coolant accident. Ductile yielding, brittle fracture and fatigue failure were considered in the analysis. The initial results of the analysis indicate that no loss of vessel integrity would be experienced even if large flaws were presumed to exist in the vessel wall at the beginning of the quenching. However, in view of the uncertainties associated with the analytical methods used to arrive at these calculated results, we believe the applicant should continue his work on this problem as expeditiously as practical. The applicant should also explore possible changes to the facility to eliminate the need to consider potential reactor vessel failure due to thermal shock.

As recommended by the report of the ACRS (Section 11), we will continue to review subsequent thermal shock information to ensure that conservative assumptions have been made and that the calculational models are supported by experimental information should this be necessary.

3.7.2 Once-Through Steam Generator

In the B&W single pass or once-through steam generator design the primary water enters the top of the steam generator, is cooled while passing downward through the Inconel tubes and exits from the bottom head. The secondary feedwater is sprayed into an annulus near the steam generator carbon-steel shell. The feedwater is heated by steam which is allowed to bypass from the heated region back to the annulus. When the feedwater reaches the bottom of the annulus it is near the saturation temperature and is boiled as it passes upward through baffling around the tubes which contain the primary fluid. When the steam exits from the generator, all the water has been evaporated and the steam is dry with about 35° F of superheat.

At full power the feedwater to the steam generator is controlled by a combination of power demand, system frequency and secondary steam pressure.

In addition to these parameters, maximum and minimum demand limits and a rate limit control the feedwater flow. This integrated controller is similar in concept to the controllers used on conventional steam plants and will be further reviewed at the operating license stage.

Feedwater quality is maintained at a high level by a polishing demineralizer. Makeup water to the secondary system will be treated by a separate demineralizer. High quality water will minimize stress-corrosion problems in the steam generators.

Since the tubes are welded to the tube sheets which are in turn fixed to the generator shell, differential expansion and stresses can be experienced when the tube and shell temperatures are different. During startup and shutdown when the temperature difference is greatest (about 40° F) the stresses are compressive and small, only about 25% of the code allowable stress for the Inconel material. Buckling of the tubes is avoided by lateral support at 40-inch intervals.

A development program for the steam generator has been proposed by the applicant, including vibration and blowdown tests, and we will require a report of the test data and an analysis of their significance before final approval of the design at the operating license stage. The applicant has indicated that both primary and secondary side blowdown tests will be performed. Our analysis to date indicates that the applicant has an acceptable design basis for the steam generators.

3.8 Secondary System

Steam passes from the steam generator at about 570° F and 910 psig through steam lines through the containment wall and to the turbine building. Safety valves are mounted on each line outside the containment. The steam passes through turbine stop valves and control valves to the turbine steam chest. After passing through the turbine, the low energy steam is condensed in the main condenser and returned through feedwater heaters and two half-capacity steam turbine-driven feedwater pumps to the steam generator. Two emergency feedwater pumps, one steam-driven and one electric-powered are provided for decay heat removal during normal or emergency shutdown. The electric-powered pump is operable from the emergency diesels.

The applicant has presented an analysis on the effects to the plant of a complete loss of AC electric power. Plants on saline coastal sites characteristically have limited capability for remaining at hot, standby conditions following loss of electric power. The plant technical specifications will have a provision that requires a plant cooldown and depressurization when a minimum feedwater inventory is reached. The minimum value, to be set at the operating license stage, is required for the assurance that the plant can be depressurized before exhaustion of feedwater.

The secondary system is designed to reduce load automatically to station auxiliary loads in case of a blackout or other transient on the external power grid. This would be accomplished by briefly venting secondary steam to the atmosphere while feedwater flow is reduced to the generators.

We believe that the proposed design of the secondary system is acceptable.

4.0 CONTAINMENT

4.1 Description

The containment structure proposed is constructed of prestressed concrete. The containment encloses the primary system, steam generators, and related auxiliaries, and is a vertical right cylinder with a shallow ellipsoidal sector dome and a flat slab base. A welded steel liner, three-eighths inch thick for the cylinder and dome and one-fourth inch thick for the base slab is attached to the inside surface of the concrete shell to provide leaktightness.

The cylinder walls are prestressed circumferentially against hoop stress by three staggered systems of prestressing tendons anchored at six vertical buttresses. The cylinder walls are also prestressed vertically with a series of uniformly spaced tendons extending from the top of the ring girder (thickened section at cylinder-dome intersection) to the bottom of the base slab. The dome is prestressed by a three way tendon system extending across the dome and anchored on a horizontal plane on the dome ring girder. The prestress tendon design will be the BBRV system, previously approved for the Metropolitan Edison Company's Three Mile Island plant and the Duke Power Company's Oconee plants.

Local base moments are carried by reinforcing bars which extend diagonally through the thickness of the slab and up the cylindrical wall about 12 feet. A grid of supplemental reinforcing bars is provided on the exterior face of the cylinder and dome for crack control. Additional reinforcement is provided at critical points on the interior face at the dome liner and in the anchorage zones. Rigid shear "T" and "L" connectors are provided on the liner exterior face to fasten the liner into the concrete.

The prestressing tendon pattern is deflected around the major cylinder penetrations (personnel and equipment access hatches) and additional steel reinforcement is provided for local moments and shear stresses.

The major loadings considered by the applicant include dead load, accident pressure, accident temperature, seismic, and wind. The applicant has also considered external pressure, tornado and missile loadings. The manner of load combination for the containment considers all significant loads and has been found acceptable by our structural design consultant, Dr. Newmark (Appendix F).

The static load stresses and deflections that are in a thin, elastic shell of revolution are calculated by an exact numerical solution of the general bending theory of shells. The equations used take into account the bending as well as membrane action of the shell.

The equipment access hatch is approximately 18 - 1/3 feet in diameter and the personnel hatch is 9-1/2 feet in diameter. The Franklin Institute has been engaged to make a computerized finite element analysis of the design of these openings. The results of this design analysis will be used as the basis for developing a confirmatory instrumentation program for the proof test. The preliminary design as described by the applicant is acceptable and we believe that the structural criteria can be met. We intend to give careful attention to the final design during the operating license review.

4.2 Construction and In-Service Surveillance

The materials of construction, i.e., the prestressing system, tendon protective coating, concrete, reinforcing steel, and liner plate materials are quality, proven materials. A retaining wall and drainage system around the Reactor Building will provide protection of the liner tendons against ground corrosion. The tendons are also enclosed in a metallic wax-filled tube for additional protection. A liberal concrete cover allowance on reinforcing steel has been specified to provide assurance that deterioration of the structure during its operating life will not be significant. All metallic components including liner plate and tendon conduits will be electrically connected to provent stray current corrosion.

The Florida Power Corporation organization will be responsible for quality control (Sec. 8.2) to ensure that the plant is constructed in accord with the requirements of the design. Independent testing agencies will be retained by Florida Power Corporation for assistance in the quality control program. Gilbert Associates will prepare final evaluation recommendations for use by the FPC Project Management for decisions on accepting or rejecting work or materials. The FPC Project Management through its Nuclear Project Manager, Construction Manager, and Quality Control Supervisor has the necessary responsibility and authority to implement the quality control procedures.

We find the proposed methods for construction, including quality control surveillance, provide assurance that a high-quality structure will be obtained.

An extensive program of acceptance testing for both structural capability and leaktightness has been indicated. The program to establish structural acceptance will include instrumentation on the equipment hatch, at the discontinuities, on selected reinforcing bars and on the liner to provide assurance that any anomalous structural behavior will be detected. Likewise, extensive preoperational integrated leakage tests are proposed to establish the structure's leakage characteristics.

Detailed in-service surveillance programs have not been established. However, the design has been changed from grouted tendons to an organic packing to provide capability for tendon retensioning, removal and replacement. We are satisfied that the structure will have adequate capability for a suitable surveillance program and review of the details of this program can be left for the operating stage review.

4.3 Seismic Design

The applicant has proposed to base the seismic design of the containment building on assumed ground accelerations of 0.05g for the design, and 0.10g for the maximum earthquake. The response spectrum proposed is a composite of the scaled El Centro recorded spectra and an analytic spectra at larger periods and is satisfactory.

The applicant has proposed design criteria for Class I components, systems, and structures (i.e. those whose failure might cause or increase the severity of a loss-of-coolant accident or result in an uncontrolled release of excessive amount of radioactivity, and those which are vital to safe shutdown and isolation of the reactor). The Class I criteria are that

a) Primary steady-state stresses combined with seismic stresses for the "design earthquake" shall be maintained within appropriate allowable code limits.

b) Primary steady-state stresses combined with seismic stresses for the "maximum probable earthquake" shall be limited so that safe and orderly shutdown is not impaired.

The containment design as proposed has a high degree of conservatism. It is concluded that the design, as presently proposed, and the construction, as indicated, will result in structures adequate for the intended purpose.

4.4 Containment Leakage Prevention

The containment leak rate is specified at 0.25% per day at 55 psig. Lines which penetrate the containment have provision for isolation. The degree of isolation redundancy depends on the function and configuration of each system. In general, lines which are (1) connected to the primary system, (2) normally open to the containment atmosphere, or (3) open to the containment as a result of an accident, are protected by redundant automatic valves. Lines which must remain open to allow functioning of engineered safety features during an accident must have provision for manually actuated isolation.

Lines which vent the containment atmosphere are closed both on signals which actuate other engineered safety features and on a high containment radiation signal. Closed systems which have a low probability of rupture during an accident are provided with at least one automatic valve external to the containment. The isolation system, including instrumentation, is designed so that no single failure can preclude containment isolation.

We have reviewed the instrumentation and valve arrangements proposed and have found that they conform to the design criteria and are acceptable.

4.5 Containment Design Pressure Analysis

A parametric analysis has been performed by the applicant to establish the peak containment pressures during a loss-of-coolant accident and to size the containment cooling systems (Sec. 6.2). A spectrum of primary system pipe break sizes between 0.4 ft^2 and 14.1 ft^2 has been evaluated to determine the response of the reactor building pressure.

The highest blowdown pressure peak (52.1 psig at 40 seconds) was found to result from a 3 ft^2 break. The highest post blowdown pressure (52.0 psig at 180 seconds) resulted from the 14.1 ft^2 break. The second pressure peak is a consequence of the assumed transfer of decay and metal-water reaction heat to the containment and is limited by the operation of the containment cooling systems. The calculated peak pressures are below the containment design pressure of 55 psig.

An analysis was also performed by the applicant to illustrate that the containment will withstand the metal-water reaction (including hydrogen recombination) associated with inoperability of core flooding systems. This calculation also gave a peak pressure of 53.2 psig, less than the containment design pressure.

Additional calculations show that the containment could withstand a metal-water reaction equivalent to 75% of the core zirconium, spread linearly over 1000 seconds, with all cooling equipment functioning. When only the coolers are operating the allowable zirconium percentage is reduced to slightly over 30% of the core. When both cooling systems are functioning with a single active failure in each system the allowable percentage is slightly over 40%. In each of the latter cases the reaction is assumed to occur linearly over 1000 seconds. Our evaluation of the containment design pressure analysis and the containment's capability to withstand metal-water reaction indicates that it is acceptable.

5.0 ELECTRICAL SYSTEMS

Incoming power will be provided by four 230 kv lines terminating at the site substation in a conventional "breaker-and-a-half" arrangement. This permits flexibility in cross-connecting the four lines to the start-up transformer, and in isolating faults. In addition, Crystal River Units 1 and 2 (fossil-fired) feed the same 230 kv substation and can supply power to Unit 3.

In order to assure core protection in the event of a loss-of-coolant accident, electrically-powered safety features must operate very soon after a postulated primary system failure. A time of 30 seconds after initiation of an accident for commencement of injection of water is a reasonable maximum time to assume in assessing the adequacy of the power system. Considering this, the ACRS (Sec. 11.0) has recommended that the proposed off-site power system should be modified to fulfill General Design Criterion 39 so that no single failure will prevent the operation of minimum electrically-powered safety features necessary to protect the core. This recommendation could be fulfilled by a second 230-kv startup transformer, or an additional transformer for the engineered safety features alone, or by some means of back-feeding power from the station generator 500 kv output transformer. We believe that the offsite power modification proposed by the ACRS can and should be implemented. The applicant has agreed to provide a second connection of offsite power to the engineered safety features busses. An existing transformer serving the fossil Units 1 and 2 will be used. This design modification is acceptable to us; the final design details will be reviewed at the operating license stage.

Onsite AC power will be provided from two 2850-kw diesel generators. Either diesel generator can supply the power required by the emergency core cooling and containment cooling systems in the event of a loss-of-coolant accident. Based on our review of the information submitted, we believe that the onsite AC power will meet a single failure criterion and is acceptable.

The station DC system consists of two independent 250/125 volt sources which provide power for DC pump motors, control and instrumentation. Each DC system will be supplied by a battery and battery chargers. Two circuits will be provided to permit one battery to back up the other. A backup charger will be provided for each battery. We believe that the proposed DC system is acceptable.

6.0 ENGINEERED SAFETY FEATURES

6.1 Emergency Core Cooling Systems

The applicant's design basis for the emergency core cooling systems is to limit the temperature transient below clad melting for the entire spectrum of reactor coolant system failures. To provide assurance that this criterion is met and to prevent any mechanical damage that might interfere with core cooling, the applicant has sized the emergency core cooling systems to limit the clad temperature transient to 2300° F or less. The calculated peak clad temperature, about 1950° F (zirconium melting temperature is 3360° F), occurs for the largest (14.1 ft^2) hot-leg break.

The applicant's criterion for maintenance of mechanical integrity during the blowdown is that deformation of reactor internals shall be limited to values which will ensure that the control rods can be inserted and that the core will be cooled. The applicant's nuclear steam supply contractor (B&W) has performed calculations to establish preliminary combinations and stress and deformation limits for the combined accident and earthquake loadings. The applicant's design basis for maintenance of mechanical integrity during blowdown are acceptable. We will review the calculational procedures including final loadings for the combined forces effects when they become available.

Core cooling for every location and size of primary coolant pipe break up to and including the double-ended rupture of a recirculation pipe will be provided by core flooding tanks (two provided), low pressure injection pumps (two provided), and high pressure injection pumps (three provided, one of which is a spare).

The core flooding tank system is composed of two tanks, each separated from the primary system by two check valves and one normally-open block valve. Borated coolant is maintained in the tanks at 600 psi by compressed nitrogen gas. Injection of the borated coolant into the primary system is initiated automatically by the stored gas energy when the reactor pressure drops below 600 psi. The tanks discharge directly to the reactor vessel. The water flows between the reactor vessel wall and the thermal shield and enters the bottom of the core. The combined coolant content of the two tanks is more than sufficient to cover the midplane of the core assuming no liquid is initially in the reactor vessel. The design values chosen for the flooding system together with conservative calculations for residual water in the pressure vessel are calculated to accomplish covering 80% of the core within 25 seconds after the double-ended rupture of a 36-inch reactor outlet (largest) line. The hot-spot clad temperature is then limited to about 1950° F.

In addition to the flooding tanks, emergency coolant injection is also provided by either of two low pressure pumps which each will deliver about 3000 gpm to the vessel at a pressure of 100 psig. These pumps initially take suction from the 350,000-gallon borated water storage tank and are converted to a recirculation mode from the reactor building sump by operator action after about 30 minutes, the time depending on the number of pumps in operation. The low-pressure injection system delivers water to the same nozzles as the core flooding tanks. Under normal shutdown conditions these pumps serve as decay heat removal pumps.

During the period while the injection water source is the borated water storage tank, the two high-pressure injection pumps in service deliver water to the four reactor inlet coolant lines. Each high pressure pump will deliver about 350 gpm at 2200 psi (operating pressure), 400 gpm at 1800 psig (safety system initiation pressure). These pumps would provide makeup for small breaks for which the reactor would remain at a high pressure. In the unlikely case that reactor pressure should remain high over a long period of time after a break so that the low pressure injection pumps could not inject directly into the vessel, the high pressure pumps could take suction from the outlet of the low pressure pump, heat-exchanger complex in the recirculation mode.

One of the three high-pressure pumps will be used continuously during plant operation to provide seal water to the reactor coolant pumps and control rod drives. The normal use of one pump provides additional assurance that at least one operable pump will be available if required for emergency service.

The applicant revised the originally-proposed emergency core cooling systems to comply with our interpretation of Criterion No. 44 of the proposed 70 General Design Criteria. Two separable core cooling systems for the recirculation mode have now been proposed, either of which can perform the core cooling function. Means are provided to detect and isolate a passive failure in one system without impairing the ability of the remaining system to deliver water to the reactor core.

We believe that, since the core flooding tanks and borated water storage tanks are passive components, and in use for only a short period of time, redundancy is not required in these systems. We conclude that the proposed emergency core cooling systems design meet the intent of the Commission's proposed General Design Criteria which were published for comment July 11, 1967. We will continue our review of the emergency core cooling as further information becomes available (see also Section 10.0, Research and Development).

The applicant has recognized that a water seal in the cold leg of the primary coolant system after a loss-of-coolant accident could lead to formation of a "steam bubble" or vapor lock above the core which might prevent core flooding. Because of this phenomenon, 14-inch diameter check valves have been proposed to relieve pressure from the hot leg to the cold leg. These would be mounted above the core in the core support barrel and would be held shut by the 30 psi differential during normal operation but would open on less than 1 psi applied in the reverse direction.

The applicant has proposed design features (such as a captured hinge design) to prevent loss of a valve during operation and has also analyzed the consequences of loss of one or more valve covers. The analysis indicated that a satisfactory DNB ratio (>1.3) would be maintained at normal power levels with one cover off but that at the 114% overpower condition (the highest thermal power calculated in any operational transient) the DNB ratio would be 1.24, or below the design value of 1.3. The applicant indicates loss of more than one valve would be detectable by a change in flow rate of about 2%. Additional information will be required at the operating license review to verify that the flow detection system can actually detect a 2% flow change. We believe that the undetected loss of a valve or valves should not lead to a DNB ratio of less than 1.3 at the overpower condition and the final design of the core is expected to meet this requirement. The applicant has stated that flow distribution studies will be made on a model of the reactor to simulate the loss of check valves.

The applicant has also considered the effects of expected vibrations in unseating the valve during normal operation and has stated that the energy imparted to the valve from the flowing water will not be great enough to induce vibration. However, in view of experience in which unexpected vibrations have occurred, we consider that it is necessary to conduct an experimental program to determine whether the valves could be unseated by induced vibrations. The applicant will conduct an experimental program to determine the vibrational characteristics of a prototype check valve and its support structure.

The check valves in the core barrel should provide a satisfactory solution to the steam bubble problem, subject to realization of the R&D objectives outlined in Section 10.0.

6.2 Containment Cooling Systems

Two containment cooling systems, of different design principles, are provided: (1) containment spray pumps which take water initially from the borated water storage tank and then from the containment pump and deliver it to the containment atmosphere through redundant spray headers and (2) three emergency cooling units each consisting of a fan and tube cooler which will remove heat from the containment atmosphere and transfer it to the low-pressure nuclear services cooling water system.

The containment cooling requirement is that the post-blowdown reactor building pressure be maintained below the design containment pressure. This requires an initial heat removal capacity of about 240 x 10^6 Btu/hr. This requirement can be satisfied by either: (1) 2 of 2 spray pumps, (2) 3 of 3 fan coolers or (3) 2 of 3 fan coolers and 1 of 2 spray pumps. Adequate containment cooling is supplied if either system is assumed to be completely inoperative or if each system is degraded by a single failure (reference Section 4.5). These systems provide adequate redundancy for containment cooling and have sufficient capacity to reduce the containment pressure (and thereby reduce leakage) after the design basis accident.

6.3 Iodine Removal System

An iodine "fixing" additive will be mixed with the containment spray water to remove iodine from the containment atmosphere after a loss-of-coolant accident. Two sprays are provided and either spray has the design capability to remove sufficient iodine from the containment atmosphere to reduce potential doses at the site boundary to Part 100 limits, or less.

As discussed in Section 9.5 of this report, without iodine reduction the exclusion boundary 2-hour dose and the low population distance total dose exceed Part 100 guidelines by factors of 3.3 and 1.6 respectively, for TID-14844 release assumptions and the proposed leak rate at 0.25%/day. The spray system with additive is proposed to bring the design basis loss-of-coolant accident doses within Part 100 guidelines.

The applicant has selected sodium thiosulfate as the additive. However, the research programs also include alternate chemical solutions. There is evidence that a significant removal of iodine from the containment atmosphere can be obtained by using a spray with a "fixing" additive. The additive changes the absorption process from liquid-mass-transfer limited to gas-phase-limited by providing an efficient "sink" within the droplet. While the removal factors needed to meet site guidelines appear to be available under laboratory conditions, the stability and compatibility of the additives under accident conditions have not yet been proven.

The applicant has outlined a research and development program designed to provide adequate information to justify the use of a chemical spray as an engineered safety feature. The program relies on current and future experiments by Oak Ridge National Laboratory (ORNL) to justify spray removal rates. B&W will study the radiation and thermal stability problem and corrosion and chemical attack on containment materials but is committed to investigate removal rates if the ORNL work is not forthcoming.

The efficiency of the chemical spray will be experimentally checked in an environment in which the spray water is hotter than the atmosphere to confirm the analytical calculations which indicate no significant decrease in efficiency under these conditions. The condition of containment atmosphere cooler than spray water could occur about 15 minutes after the postulated accident as a result of operation of the building coolers.

The research and development programs outlined by the applicant in conjunction with current studies at ORNL should provide a satisfactory iodine removal system at the operating stage. Also, these programs should show that the required reduction factors, on the order of 3.3 in 2 hours and 1.6 over the course of the accident, can be achieved or exceeded. Should the research and development programs show spray systems to be unacceptable for iodine removal, alternate means to reduce dose rates will be employed. Charcoal filters and a reduced leak rate are among the alternates that could be approved.

7.0 RADIOACTIVE WASTE CONTROL

The waste handling and storage equipment has been sized to accommodate continued reactor operation with clad defects in 1% of the fuel rods. The primary system is maintained at high water purity and radioactive wastes removed by the chemical purification system. A small stream is bled from the primary system, reduced in pressure and temperature by the letdown coolers and passed through a demineralizer as necessary and then routed to the letdown storage tank. Makeup to the primary system is provided by pumping the water from the letdown storage tank through the seal water or high pressure injection system. Addition or dilution of borated water is also accomplished by this system by feeding the letdown storage tank from the chemical addition system.

Liquid wastes are collected from the demineralizer sluice or other miscellaneous sources, concentrated in evaporators and packaged for offsite disposal. Low concentration condensate from the evaporators is either reclaimed for reuse in the primary system, or is diluted to concentrations below those specified in 10 CFR Part 20 prior to release to the salt-water discharge canal.

Solid wastes will be stored temporarily pending shipment from the site in containers approved for the purpose.

Gaseous wastes will be monitored, diluted and released to the atmosphere or stored in waste gas holdup tanks to provide for appropriate decay of the radioactivity.

The applicant has proposed a monitoring system for all likely sources of effluent release (as discussed in Section 3.6.4 of this report) and has performed an analysis on the liquid waste disposal systems to show that multiple equipment failures and operator errrors would be required to allow undetected discharge of radioactive wastes.

The waste disposal system described by the applicant will effectively control radioactive wastes generated on the site and it is therefore acceptable. The proposed release limits for the site will be reviewed by the regulatory staff at the operating license review stage.

8.0 TECHNICAL QUALIFICATIONS

8.1 Organization

The Production Department of the Florida Power Corporation, headed by the Production Superintendent, is responsible for all electric generating plant operations. The Crystal River plant superintendent reports to the Production Superintendent who in turn reports to the Vice-President-Power. The Mechanical Engineering Department, also reporting to Vice-President-Power, provides technical support in the areas of mechanical, electrical, control, and architectural-structural engineering. The Chief Mechanical Engineer is presently also the Nuclear Project Manager and in this latter role reports directly to the company president. A construction section headed by the Construction Manager is part of the Mechanical Engineering Department and will supervise and coordinate the construction of the nuclear units. This section has experience in power plant construction management, having exercised this supervision on all major plants in the FPC system.

The number of people proposed for operation of the Nuclear Unit 3 totals 59. This includes supervision, maintenance, custodian, nuclear control, electrical control, and technical support. All of the supervisory positions will be filled with men with extensive operating and maintenance experience in fossil-fueled steam-electric generating plants, and who will be given nuclear training.

Training for the FPC staff will include selected college courses, special nuclear engineering courses conducted by B&W, and short courses in reactor operation instruction at the B&W facilities at Lynchburg, Virginia. In addition, operator trainees will spend 6 months in residence at an operating nuclear plant. B&W will assist in on-the-job training and operator licensing preparations, and will work with the operating staff throughout loading and startup testing.

Florida Power Corporation will act as construction manager; the FPC Nuclear Project Manager, reporting directly to the FPC president, has final authority for design and construction decisions. The B&W Company has extensive background in supplying nuclear steam supply systems. Gilbert Associates, has been associated with nuclear project designs since 1955, including the Metropolitan Edison Three Mile Island and the Rochester Gas & Electric Ginna plant. The J. A. Jones Company, general contractor, has worked for the Atomic Energy Commission in building the gaseous diffusion plant at Oak Ridge and a major part of the Hanford Plant.

On the basis of the above considerations, we conclude that the applicant and its contractors, B&W, Gilbert Associates, and J. A. Jones Company are suitably qualified to design and construct the proposed facility.

8.2 Quality Control

Quality Control during construction will be coordinated by a Quality Control supervisor in the employ of Florida Power Corporation. The Quality Control supervisor will be responsible to the Construction Manager for quality assurance and the quality control program. Independent testing agencies will be retained as necessary.

All reports or evaluations from consulting agencies will be submitted to the Quality Control supervisor. The Quality Control supervisor normally reports in a staff function to the Construction Manager. However in all cases of violation or non-compliance with codes or standards the Quality Control supervisor has direct access to the Florida Power Corporation Nuclear Project Manager who will resolve any differences.

Gilbert Associates will prepare quality control procedures for all elements of construction other than the nuclear steam supply system. This includes detailed specifications, testing procedures, and surveillance of tests during construction. All test reports, including those for steel, cement, and welding procedures and qualifications, will be reviewed by Gilbert Associates. The Gilbert resident engineer will recommend work stoppage through the FPC management if necessary to achieve design objective. Gilbert Associates will also provide personnel for shop inspection as required by Florida Power Corporation.

Babcock & Wilcox Company will be responsible for quality of workmanship for both B&W manufacturing and field assembly functions. B&W personnel will perform quality control inspection functions during the performance of all welding processes. Upon assembly, systems inspection will be performed to assure adequacy of cleaning techniques prior to operational test.

Florida Power Corporation, through the above lines of internal responsibility and through contractual agreements, has the final authority for rejection of materials and stoppage of work as may be required by the quality assurance program. All records and documentation required by the quality assurance program will be received and maintained by FPC personnel.

We conclude that the quality assurance and control program proposed by Florida Power Corporation is acceptable.

9.0 ACCIDENT ANALYSIS

9.1 Incidents

A number of operational transients were considered by the applicant including control assembly withdrawal during startup and from power, moderator dilution, and loss-of-coolant flow. The applicant's evaluation concluded, and we agree, that no significant radiological hazard would result. The Babcock & Wilcox Company is pursuing a research program to gain knowledge of physical fuel properties at high burnups which should provide knowledge concerning the ability of the fuel to withstand expected transients at the end of its design lifetime.

A number of incidents were evaluated including a waste gas tank failure and an accidental release of liquid effluent. An accidental discharge of 450 gallons of liquid waste (evaporator condensate) at activity levels corresponding to continued operation with 1% failed fuel was postulated to occur over a 1-hour period and be diluted by the service water system effluent. Multiple equipment failures and operator errors would be required before the radioactive effluent could be released. The calculations show that accidental discharge of these operational stored wastes would result in concentrations well below 10 CFR Part 20 limits in the condenser discharge canal. The release of activity from a waste gas tank relief valve failure after operation with 1% failed fuel was calculated to be within 10 CFR Part 20 limits.

9.2 Steam Generator Tube Rupture

The double-ended rupture of a steam generator tube was postulated. The radiological consequences were calculated based on prior operation with 1% failed fuel, primary to secondary leak rate of 435 gpm for three hours, and concurrent loss of offsite power. Under these conditions the iodine in the primary system coolant is released to the secondary system and thence to the atmosphere. The 2-hour thyroid dose at the site boundary is approximately 100 rem, within Part 100 guidelines.

9.3 Steam Line Break

A steam line failure was analyzed which resulted in the release of the fission products contained in the secondary system (which might be accumulated due to minor tube leakage in the steam generator). The applicant stated that the releases from this accident would be small and we agree that the resultant doses would be well within the 10 CFR Part 100 guidelines.

A break in a main steam line during operation would cause cooldown of the primary system due to flashing of the secondary system inventory. The flashing of the relatively small feedwater inventory would cause a decrease in primary coolant temperature of about 50°F at the end-of-life conditions when the maximum negative moderator temperature coefficient is present. The large increment of reactivity held by the control assemblies and the injection of boron from the high pressure injection system is calculated to maintain the core in a shutdown condition even if one maximum-worth control assembly were assumed to remain out of the core.

9.4 Control Assembly Ejection Accident

The ejection of a control assembly from the core is postulated to occur as a result of a break in the pressure housing of the control rod drive. The maximum reactivity increment that could be inserted corresponds to the worth of the ejected assembly in the core prior to the accident. The maximum worth of a control assembly at full power is 0.46% delta k/k and the maximum worth at source level is 0.6% delta k/k. The parametric study presented showed the effect of ejected assemblies worth 0.1% to 0.7% delta k/k for both the full power and source level cases.

For the ejection of a 0.46% assembly from full power the maximum enthalpy in the hottest fuel rod was calculated to be about 80 calories per gram (cal/gm). The applicant's sensitivity analysis, which arbitrarily increased the worth of the ejected assembly, indicates that ejection of an assembly worth 0.6% from full power would result in a hot spot enthalpy of about 200 cal/gm. This is still below the fuel melting temperature and no significant rapid energy release to the water is expected. An ejection of a 0.5% delta k/k assembly at source power was calculated by ejecting a 1% assembly with the core initially 0.5% delta k/k subcritical. The results of the analysis indicate a resultant peak power level of about 40% full power.

A sensitivity analysis was performed to show the effect of variation of important parameters in addition to control assembly worth. These included Doppler coefficient, moderator coefficient and trip delay time. No large variations in the computed results were observed when the above parameters were varied over a range of values. The environmental analysis performed assumed that fuel gap activity was released into the containment building. The resultant doses at the site boundary would be small and well within 10 CFR Part 100 guidelines.

We believe that the results of the applicant's analyses show that vessel failure would not occur as a result of an ejected control assembly, of the worths calculated for this design, and that damage to core internals would not be expected at the peak enthalpy values calculated since they are well below the threshold for rapid energy addition to the water.

9.5 Loss-of-Coolant Accident

The applicant has proposed an emergency core cooling system (including core flooding tanks) which is designed to protect the core for the full spectrum of primary system break sizes which would result in a loss-of-coolant up to the double-ended rupture of the largest pipe in the system. The applicant has presented core cooling analyses for a spectrum of break sizes. We believe that the spectrum-of-breaks analyses completed to date provide assurance that the clad temperature transient will be limited to a value well below melting. Certain aspects of emergency core cooling are considered R&D items (Sec. 10.0).

The highest clad temperatures (about 1950°F) were calculated to occur for the largest break and the applicant has concluded that no deformations due to core heatup which could cause interference with cooling are expected at these temperatures. The ACRS has recommended (Sec. 11.0) that additional work be done to assure that fuel rod failures in loss-of-coolant accidents will not affect significantly the ability of the emergency core cooling systems to prevent clad melting. In addition to experimental and analytical work being carried out by B&W on fuel rod failure mechanisms, (Sec. 10.0) several other programs of this nature are in progress in the nuclear industry and we will follow the progress of all of these programs to assure that any significant results of the programs are incorporated in the final design.

The final calculations of blowdown forces from coolant line ruptures have not been completed. However, preliminary estimates have been supplied by the applicant, including load combinations and deformation limits (Section 3.3 of this report). The applicant has also performed a thermal shock analysis on the effect of ECCS action on the pressure vessel and internals. We conclude that these preliminary analyses are acceptable in scope and will review the final results when available. As discussed in Section 4.6 of this report, the applicant performed parametric analyses to establish the peak accident pressure in the containment. We believe that the analyses are acceptable.

The applicant has calculated the environmental consequences of the loss-of-coolant accident for the expected course of the accident and for a "design basis accident" in which 100% of the noble gases and 50% of the halogens and 1% of the particulates are assumed to be released to the reactor containment where 50% of the halogens were assumed to plate out. (This corresponds to assumptions recommended in TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites.") The reactor building leak rate was assumed constant at the design leak rate of 0.25%/day for the first 24 hours and then was assumed constant at 0.125%/day for the duration of the accident. The assumed meteorology during the accident uses the Pasquill dispersion model, modified to account for additional dispersion due to the containment building wake. Doses are computed along the plume centerline for the first 24 hours. Sector averaging over a 22-1/2° arc was performed for the next 29 days. Conservative estimates of wind direction persistence and meteorological stability classes were assumed. We find the proposed accident meteorology conservative and acceptable.

The applicant assumed iodine removal by the sodium thiosulfate sprays with a removal constant of $\lambda = 25.3\ hr^{-1}$ which would result in removal of one-half the iodine in about 1.6 minutes. Five percent of the iodine inventory was assumed to be in a nonremovable form and this fraction accounts for most of the applicant's calculated 2-hour dose of 65 rem at the 4400-foot exclusion boundary and 38 rem at the low population distance of 5 miles in 30 days. With this high removal constant for iodine the dose becomes directly proportional to the amount assumed nonremovable and to the atmospheric dilution assumed.

The applicant performed a parametric study to demonstrate the effect of a lower than expected iodine removal rate. The applicant has stated that Part 100 guidelines could be met with only one chemical spray system operating and a drop size of twice that expected.

It should be noted that there is some experimental evidence that organic forms of iodine, which are assumed nonremovable in the calculation, are also removed by the spray but at a slower rate. This could further reduce the calculated course-of-the-accident thyroid doses.

We have also calculated the potential doses from this accident assuming that both sprays were not operable to determine the iodine removal factors which must be achieved to meet 10 CFR Part 100 guidelines for thyroid doses. Our calculations indicate that (1) the 2-hour thyroid dose at the exclusion boundary of 4400 feet would be a factor of 3.3 higher than the 300 rem guideline dose and (2) the course-of-the-accident thyroid dose at the low population distance of 5 miles would be a factor of 1.6 higher than the 300 rem guideline dose if the sprays were inoperable. As discussed in

Section 6.3 of this report, we believe that the experimental work performed to date and the research and development program outlined by the applicant provide reasonable assurance that reduction factors on the order of those described above can be achieved.

The whole body doses calculated by the applicant from a passing cloud are 1.9 rem for the 2-hour dose at the exclusion area boundary and 1.2 rem for the course-of-the-accident dose at the low population distance. Using somewhat more conservative assumptions our calculations indicate the doses would be slightly less than 5 rem.

With the engineered safety features proposed, the reactor facility conforms to the Commission's guidelines given in 10 CFR Part 100 and therefore is acceptable.

10.0 RESEARCH AND DEVELOPMENT

The applicant has identified a number of areas in which research and development will be carried out, primarily by B&W. We will follow the research and development programs identified below by meeting with the applicant and its contractors and by evaluating reports submitted on these programs as they are submitted. Those areas involving research and development programs are:

(1) Once-through steam generator

Steady-state conditions and operational transients will be investigated in conjunction with the control system to be used. Vibration tests, including steam generator response to primary system blowdown, will be investigated and the thermal response to both primary and secondary blowdown determined.

(2) Control rod drive unit test

The prototype tests (Section 3.5) outlined by the applicant to be conducted under operating temperature, pressure, flow and water chemistry should provide information on the operability and reliability of the system.

(3) In-core neutron detectors

The self-powered in-core neutron detector units are currently under test in the Big Rock Point Nuclear Power Plant. The status of the tests to date and the plans for completing the development are acceptable.

(4) Thermal and Hydraulic Programs

The applicant has proposed scaled flow distribution tests on the vessel and internals and rod bundle tests to determine local mixing and flow effects as discussed in Section 3.4 of this report. This further experimental and analytical work must be done to determine the limiting heat fluxes at various positions within the fuel bundle if the design is to be based on the B&W heat transfer data.

(5) Core Cooling

The core cooling research and development must specifically include (a) the completion of the analysis of the spectrum of small break sizes in the loss-of-coolant accident, (b) the development of the analytical techniques for determining blowdown forces on reactor internals, and (c) demonstration that the injection coolant will cool the core including consideration of core bypass or formation of a vapor lock. As discussed in Section 6.1, experimental vibration tests will also be performed to show that induced-vibrations will not unseat the core barrel check valves.

(6) Xenon oscillations

The applicant will further develop analytical techniques to determine whether xenon oscillations can occur. If oscillations are possible a system for controlling the oscillations will also have to be developed.

(7) Iodine Removal System

The R&D program includes studies on decomposition under normal and accident conditions, materials compatibility, iodine removal characteristics, and compatibility with boron compounds. Also included will be parallel tests on alternate chemical solutions, and tests on spray efficiency during conditions of spray water hotter than ambient atmosphere (Section 6.3).

(8) Fuel rod failure mechanisms during LOCA

Various failure modes of the fuel rods during the LOCA, such as clad melting, eutectic formation, bulging, splitting, or brittle failure, will be examined in an experimental program to assure the continued core cooling capability during a LOCA. We consider the proposed program to be satisfactory.

We conclude that the research and development programs described above provide reasonable assurance that the respective safety questions can be resolved at the operating license review stage.

11.0 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Advisory Committee on Reactor Safeguards, by letter to Chairman Seaborg dated May 15, 1968, reported on the Crystal River Unit 3 Nuclear Generating Station. A copy of this report is attached as Appendix A. The report contained comments and recommendations which we are implementing as noted in the appropriate sections of this report. The items mentioned will be resolved, prior to the issuance of an operating license, to the satisfaction of the staff and the ACRS.

In its letter on Crystal River the ACRS referred to previous letters on the Three Mile Island and Oconee plants, and concluded that matters in those letters applicable to all large, water-cooled, power reactors apply similarly to Crystal River.

The items in the Crystal River report, and the applicable items in the Three Mile Island and Oconee letters are listed below, together with a reference page in this safety evaluation.

1. Modification of offsite electrical power system (page 36).
2. Three Mile Island ACRS report.
    a. Diversity of actuation signal for ECCS (page 21).
    b. Split scram bus (page 21).
    c. Separation of control and safety (page 20).
    d. Failed fuel-element detection (page 20).
    e. Fuel rod failures during loss-of-coolant accidents (page 60).
    f. Part-length rods for xenon oscillation control (page 13).
    g. Control of positive moderator coefficient (page 12).

h. Effects of blowdown forces (page 13).

i. Effects of thermal shock induced by ECCS action (page 55).

j. Core barrel check valve vibration studies (page 42).

3. Oconee ACRS Letter.

a. Spectrum of breaks analysis (page 38).

b. Thermal shock (see 2-i, above).

c. Blowdown forces (see 2-h above).

d. Fuel rod failures during loss-of-coolant accidents (see 2-e above).

e. Core barrel check valve review (page 41).

f. Diversity of actuation signal by ECCS (see 2-a above).

g. Primary system quality assurance and inspection (page 50).

h. Control of positive moderator coefficient (see 2-g above).

i. Fuel rod transients at end of core lifetime (page 51).

j. Stability margin for xenon oscillation (see 2-f above).

The report concluded: "The Advisory Committee on Reactor Safeguards believes that, if due consideration is given to the foregoing items, the proposed reactor can be constructed at the Crystal River site with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

12.0 CONFORMANCE TO THE GENERAL DESIGN CRITERIA

In November 1965, the Commission published its General Design Criteria for Nuclear Power Plant Construction Permits. In the PSAR, the applicant evaluated the unit considering these criteria. However, on July 11, 1967, the Commission published in the Federal Register its revised General Design Criteria taking into account comments received on the initial criteria and further development of the criteria by the regulatory staff.

In response to our request the applicant evaluated the proposed design against the revised criteria, and concluded that the facility will be designed to meet the intent of the criteria. At present the proposed facility conforms to the intent of the revised criteria. Recognizing that the proposed revised criteria may be modified as a result of comments by interested parties, we intend to review the proposed unit at the operating license stage in light of the criteria as formulated at that time.

13.0 COMMON DEFENSE AND SECURITY

The application reflects that the activities to be conducted would be within the jurisdiction of the United States and that all of the directors and principal officers of the applicant are American citizens. We find nothing in the application to suggest that the applicant is owned, controlled or dominated by an alien, a foreign corporation or a foreign Government. The activities to be conducted do not involve any restricted data, but the applicant has agreed to safeguard any such data which might become involved in accordance with paragraph 50.33(j) of 10 CFR Part 50. The applicant will rely upon obtaining fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material from military purposes is involved. For these reasons and in the absence of any information to the contrary, we have found that the activities to be performed will not be inimical to the common defense and security.

14.0 CONCLUSIONS

Based on the proposed design of the Florida Power Corporation's Crystal River Unit 3 Nuclear Generating Station, on the criteria, principles and design arrangements for systems and components thus far described, which include all of the important safety items, on the calculated potential consequences of routine and accidental release of radioactive materials to the environs, on the scope of the development program which will be conducted, and on the technical competence of the applicant and the principal contractors, we have concluded that, in accordance with the provisions of paragraph 50.35(a), 10 CFR Part 50 and paragraph 2.104(b) 10 CFR Part 2:

1. The applicant has described the proposed design of the facility, including the principal architectural and engineering criteria for the design and has identified the major features or components for the protection of the health and safety of the public;
2. Such further technical or design information as may be required to complete the safety analysis and which can reasonably be left for later consideration will be supplied in the final safety analysis reports;
3. Safety features or components, which require research and development have been described by the applicant and the applicant has identified, and there will be conducted, a research and development program reasonably designed to resolve any safety questions associated with such features or components;

4. On the basis of the foregoing, there is reasonable assurance that (i) such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facility and (ii) taking into consideration the site criteria contained in 10 CFR Part 100, the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public;
5. The applicant is technically qualified to design and construct the proposed facility; and
6. The issuance of a permit for the construction of the facility will not be inimical to the common defense and security or to the health and safety of the public.

APPENDIX A

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

MAY 15 1968

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

Dear Dr. Seaborg:

At the special ACRS meeting on April 27, 1968, the Advisory Committee on Reactor Safeguards reviewed the proposal of the Florida Power Corporation to construct the Crystal River Nuclear Generating Plant. This project had been previously considered at a Subcommittee meeting held on February 15, 1968, at the site. During its review, the Committee had the benefit of discussions with representatives and consultants of the Florida Power Corporation, the Babcock and Wilcox Company, Gilbert Associates, Inc., and the AEC Regulatory Staff. The Committee also had available the documents listed below.

The plant will be located on the Gulf of Mexico about 70 miles north of Tampa, Florida, and 7 1/2 miles northwest of the town of Crystal River. The population, including Crystal River, within a ten mile radius of the plant is 3300. The site comprises 4738 acres, on which the Florida Power Corporation operates a 387 MWe coal-fired Unit 1 and is building a coal-fired 510 MWe Unit 2. Unit 3 will use a pressurized water reactor, rated at 2452 MWt and 855 MWe.

The program for foundation grouting and the protection to be provided against flooding appear to be satisfactory as do other site-related factors.

The Committee believes that the proposed off-site power system should be modified to fulfill Criterion 39 so that no single failure will prevent the operation of minimum electrically-powered safety features necessary to protect the core.

The proposed Unit 3 is similar to the Duke Power Company's Oconee Units (ACRS Report, July 11, 1967) and the Metropolitan Edison Company's Three Mile Island Unit (ACRS Report, January 17, 1968). The Committee continues to call attention to matters that warrant careful consideration for all large, water-cooled, power reactors. These matters, stated in the Three Mile Island and Oconee reports, apply similarly to the Crystal River Unit 3.

The Advisory Committee on Reactor Safeguards believes that, if due consideration is given to the foregoing items, the proposed reactor can be constructed at the Crystal River site with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Mr. Harold Etherington did not participate in the Committee's review of this project.

Sincerely yours,

Carroll W. Zabel

Carroll W. Zabel
Chairman

References attached.

References - Crystal River

1. Letter from Florida Power Corporation, dated August 10, 1967; Application for License; Volumes 1, 2, 3 and 4 of Preliminary Safety Analysis Report, Crystal River Units 3 and 4 Nuclear Generating Plant

2. Letter from Florida Power Corporation, dated January 15, 1968; Amendment No. 1 to License Application

3. Letter from Florida Power Corporation, dated February 7, 1968; Amendment No. 2 to License Application; Supplement No. 1

4. Letter from Florida Power Corporation, dated March 1, 1968; Amendment No. 3 to License Application; Supplement No. 2

5. Letter from Florida Power Corporation, dated March 11, 1968; Amendment No. 4 to License Application; Supplement No. 3

6. Letter from Florida Power Corporation, dated April 1, 1968; Amendment No. 5 to License Application

APPENDIX B

Chronology of Regulatory Staff's Review of the Crystal River Unit 3 Nuclear Generating Plant

1.	August 10, 1967	Submittal of Preliminary Safety Analysis Report (for two nuclear units).
2.	September 26, 1967	Meeting with applicant to discuss staff review areas and schedule.
3.	November 30, December 1, 1967	Meeting with applicant to discuss greater detail required.
4.	January 15, 1968	Submittal of Amendment No. 1 updating application to reflect core design changes, prestress tendon cover, and including hurricane and wave protection.
5.	January 19, 1968	Questions issued to applicant requesting additional design and safety information.
6.	February 7, 1968	Submittal of Amendment No. 2; answer to staff questions of January 19, 1968.
7.	February 15, 1968	ACRS Subcommittee meeting with staff and applicant at site.
8.	February 21, 1968	Meeting with applicant to discuss information submitted in Amendment No. 2.
9.	March 4, 1968	Submittal of Amendment No. 3 to incorporate additional information on foundation and to modify control and instrumentation.
10.	March 13, 1968	Submittal of Amendment No. 4 to clarify seismic design criteria.
11.	April 8, 1968	Submittal of Amendment No. 5 to reduce plant design to only one nuclear unit.
12.	April 27, 1968	ACRS meeting to discuss technical aspects of design.
13.	May 15, 1968	ACRS Report on Crystal River Unit 3 Nuclear Generating Plant.

APPENDIX C

OPTIONAL FORM NO. 10
MAY 1962 EDITION
GSA FPMR (41 CFR) 101-11.6

UNITED STATES GOVERNMENT

Memorandum

TO : Peter A. Morris, Director
Division of Reactor Licensing

DATE: OCT 20 1967

FROM : Milton Shaw, Director
Division of Reactor Development & Technology

SUBJECT: SAFETY ANALYSIS REPORTS

RDT:NS:S288

Reference is made to the letters of August 17, 1967, September 15, 1967, and September 18 and 26, 1967, from the Division of Reactor Licensing, to the Environmental Science Services Administration requesting comments on the following safety analysis reports respectively:

Crystal River Units 3 and 4
Florida Power Corporation

Southwest Experimental Fast Oxide Reactor
Facility Description and Safety Analysis Report
Volumes I and II dated July 24, 1967

Oyster Creek Nuclear Power Plant Unit No. 1
Jersey Central Power and Light Company
Facility Description and Safety Analysis Report
Amendment No. 11 dated Aug. 30, 1967
Amendment No. 13 dated Sept. 7, 1967

Review by the Environmental Meteorology Branch, Air Resources Laboratory, ESSA, has now been completed and their comments are attached.

Attachments:
Three Sets of Comments
(Orig. & 1 copy)

Comments on

Crystal River Units 3 and 4
Florida Power Corporation

Prepared by

Environmental Meteorology Branch
Air Resources Laboratory
Environmental Science Services Administration
October 18, 1967

The meteorological conditions assumed for the accident case are conservative. For the first period (0 - 24 hours) Pasquill Type F, 1 m/sec, a ground source, and an invariant plume centerline was used, in addition to a building wake dilution effect which amounted to an extra dilution factor of 1.5 at the site boundary of 1340m. For the remaining period up to 30 days the concentrations were averaged over a 22 1/2 degree sector which amounts to about a factor of 3 at a distance of 10^3 meters as compared to an invariant plume centerline.

Since the plant is to be located 1 mile inland from the Gulf of Mexico shoreline, there probably will be very little stabilizing effect remaining on a ground source because of the air having initially traveled over a smooth water surface.

In summary, there appear to be no unique meteorological situations that have not been considered in the application. The net result of the atmospheric diffusion parameters assumed for the accident case is a conservative set of dose calculations.

APPENDIX D

UNITED STATES
DEPARTMENT OF THE INTERIOR
GEOLOGICAL SURVEY
WASHINGTON, D.C. 20242

APR 2 1968

Mr. Harold L. Price
Director of Regulation
U. S. Atomic Energy Commission
4915 St. Elmo Avenue
Bethesda, Maryland 20545

Dear Mr. Price:

Transmitted herewith in response to a request by Mr. Roger S. Boyd, is a review of geologic and hydrologic aspects of the Crystal River Unit 3 near Citrus County, Florida, proposed by the Florida Power Corporation for location of a nuclear-powered thermal electric station.

The review was prepared by H. H. Waldron and E.L. Meyer and has been discussed with members of your staff. We have no objections to your making this review a part of the public record.

Sincerely yours,

William T. Pecora

Acting Director

Enclosure

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1968 APR 3 PM 1 15
U.S. ATOMIC ENERGY COMM.
REGULATORY
MAIL & RECORDS SECTION

Crystal River Unit 3
Citrus County, Florida
AEC Docket 50-302

Hydrology

The site is located on the shore of the Gulf of Mexico between the Crystal River and the Withlacoochee River, about 70 miles north of Tampa, Florida.

The unit will draw cooling water for steam condensors from the Gulf at a rate of 2,000 cubic feet per second. The intake and discharge canals extend for some distance offshore into Crystal Bay which is shallow, ranging from 2 to 6 feet in depth and is dotted with reefs. The mean tidal range in the Bay varies from about 2 to 3 feet.

Hurricane tides will present the critical flood problem at the site. The flood protection design level is to be based on the maximum probable hurricane tides and wave run up.

Ground-water levels near the coast would be expected to slope towards the shore, and ground-water runoff would be towards the Gulf. Spills of radioactive liquids at or near the site could be expected to discharge through the ground into the Gulf.

The reactor at this location is not likely to affect the fresh water resources of the area.

Geology

The analysis of the geology of the Crystal River Nuclear Generating Plant in Florida, as presented in AEC Dockets 50-302, and -303 and supplements, was reviewed and compared with the available literature; foundation conditions and proposed treatment of the foundation were reviewed at the site on February 14 and 15, 1968. The analysis appears to be carefully derived and to present an adequate appraisal of those aspects of the geology that would be pertinent to an engineering evaluation of the site.

There are no positively identifiable active faults or other recent geologi structures that could be expected to localize earthquakes in the immediate vicinity of the site.

Tectonically the site is located on the western flank of the Ocala uplift, which is the dominant subsurface structural element in the western part of the northern peninsular structural province of Florida. Although several faults are associated with, and essentially parallel to, this northwest-trending anticlinal fold, all available evidence indicates that the structure has not been tectonically active since late Tertiary times.

Foundation rock at the site is a granular, clastic limestone (the Inglis Limestone Member of the Moodys Branch Formation) that is characterized by solution cavities and by major zones of friable, poorly cemented rock. The applicant is aware of the problems involved with such a foundation material and recognizes the need for a suitable and carefully controlled treatment of the limestone in order to assure the integrity of the rock as a foundation material.

Treatment of the foundation rock, as proposed, will be accomplished by means of curtain and consolidation grouting to specified minimum terminal unit takes of grout, utilizing the split-spacing and stage-grouting techniques. By this method of treatment, the existence of any large cavities in the limestone that underlies the plant structure should be precluded, and resulting grout-stabilized rock should provide for an adequate foundation for the proposed nuclear facility.

APPENDIX E

U.S. DEPARTMENT OF COMMERCE
ENVIRONMENTAL SCIENCE SERVICES ADMINISTRATION
COAST AND GEODETIC SURVEY
ROCKVILLE, MD. 20852

March 15, 1968

IN REPLY REFER TO: C23

Mr. Harold L. Price
Director of Regulation
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Price:

In accordance with your request, we are forwarding 10 copies of our report on the seismicity of Citrus County, Florida, and vicinity. The Coast and Geodetic Survey has reviewed and evaluated the information on the seismic activity of the area as presented by the Florida Power Corporation in the "Preliminary Safety Analysis Report," and we are now submitting our conclusions on the seismicity factors.

If we may be of further assistance to you, please do not hesitate to contact us.

Sincerely yours,

James C. Tison, Jr.
Rear Admiral, USESSA
Director

Enclosure

REPORT ON THE SITE SEISMICITY FOR THE CRYSTAL RIVER NUCLEAR PLANT, FLORIDA

At the request of the Division of Reactor Licensing of the Atomic Energy Commission, the Seismology Division of the Coast and Geodetic Survey has evaluated the seismicity of the area around the proposed reactor site near Crystal River, Citrus County, Florida, and has reviewed a similar analysis made by the applicant and presented in the "Preliminary Safety Analysis Report." The applicant's report on the site seismicity is adequate for an evaluation of the seismic factor.

In reviewing the seismicity, the dominant factor considered was the Charleston, South Carolina, earthquake of 1886. The response of the area around the proposed reactor site to this earthquake has been evaluated at intensity VI. No other seismic activity has generated a higher intensity.

Based upon the review of the site seismicity, geology and ground conditions, the Coast and Geodetic Survey recommends that an acceleration of 0.05 g would be adequate for representing earthquake disturbances likely to occur within the lifetime of the facility. In addition, the Survey recommends an acceleration of 0.10 g to represent the ground motion from the maximum earthquake likely to affect this site. It is believed that this value would provide an adequate basis for designing protection against the loss of function of components important to safety.

U. S. Coast and Geodetic Survey
Rockville, Maryland 20852

March 13, 1968

NATHAN M. NEWMARK
CONSULTING ENGINEERING SERVICES

APPENDIX F

1114 CIVIL ENGINEERING BUILDING
URBANA, ILLINOIS 61801

26 April 1968

Dr. Peter A. Morris, Director
Division of Reactor Licensing
U. S. Atomic Energy Commission
Washington, D.C. 20545

Re: Contract No. AT(49-5)-2667
Crystal River Unit 3 Nuclear Generating Plant
Florida Power Corporation (Docket No. 50-302)

Dear Dr. Morris:

This will confirm that today I signed the above-referenced report dated April 1968.

Sincerely yours,

N M Newmark

N. M. Newmark

bjw
cc: W. J. Hall

REPORT TO AEC REGULATORY STAFF

ADEQUACY OF THE STRUCTURAL CRITERIA FOR CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

FLORIDA POWER CORPORATION

(AEC Docket No. 50-302)

by

N. M. Newmark

W. J. Hall

and

A. J. Hendron, Jr.

April 1968

ADEQUACY OF THE STRUCTURAL CRITERIA FOR THE CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

BY

N. M. Newmark, W. J. Hall and A. J. Hendron, Jr.

INTRODUCTION

This report concerns the adequacy of the containment structure and components, reactor piping and reactor internals for the Crystal River Unit 3 Nuclear Generating Plant, for which application for a construction permit has been made to the U.S. Atomic Energy Commission (AEC Docket No. 50-302) by the Florida Power Corporation. The facility is to be located in the northwestern portion of Citrus County, Florida, on the Gulf of Mexico between the mouths of the Withlacoochee and Crystal Rivers, and approximately 7-1/2 miles NW of Crystal River, and 70 miles N of Tampa, Florida. Specifically this report is concerned with the evaluation of the design criteria that determine the ability of the Class I structures, systems, and components to withstand a design earthquake acting simultaneously with other applicable loads forming the basis of the design. The facility also is to be designed to withstand a maximum earthquake simultaneously with other applicable loads to the extent of insuring safe shutdown and containment. This report is based on information and criteria set forth in the Preliminary Safety Analysis Report (PSAR) and amendments thereto as listed at the end of this report. We have participated in discussions with the AEC Regulatory Staff, and the applicant and its consultants, in which many of the design criteria were discussed in detail.

DESCRIPTION OF THE FACILITY

The Crystal River Nuclear Generating Plant is described in the PSAR as a pressurized water reactor nuclear steam supply system furnished by the Babcock &

Wilcox Company and designed for an initial power output of 2452 Mwt (855 Mwe net). The reactor coolant system consists of the reactor vessel, coolant pumps, two steam generators, pressurizer, and interconnecting piping. The reactor vessel will have an inside diameter of about 14.3 ft, a height of 41.7 ft, is designed for a pressure of 2500 psig and a temperature of 650° F, and will be made of SA-302, Grade B, steel clad with type 304 austenitic stainless steel.

The containment for this plant consists of two systems as follows: (1) the reactor building which provides biological and missile shielding, and which contains the energy and material that might be released by an accident; and (2) the engineered safeguards systems which limit the maximum value of the energy released by an accident.

The reactor building, which encloses the reactor and steam generators, consists of a steel lined concrete shell in the form of a reinforced concrete vertical cylinder with a flat base and a shallow dome roof. The cylindrical structure of 130 ft inside diameter has side walls rising 157 feet from the top of the foundation slab to the spring line of the dome roof. The concrete side walls of the cylinder and dome will be approximately 3 ft 6 in. and 3 ft 0 in. in thickness, respectively. The foundation mat will be approximately 9 ft thick with a 2-foot-thick concrete slab over the bottom liner plate. The foundation slab will be reinforced with conventional steel reinforcing. The cylindrical walls will be prestressed with a post-tensioning system in the vertical and horizontal directions. The dome roof will be prestressed utilizing a three-way post-tensioning system. The inside surface of the reactor building will be lined with a carbon steel liner 3/8-inch thick for the cylinder and dome and 1/4-inch thick for the base. The reactor building design is essentially the same as for the containment buildings for the Turkey Point, Oconee, and Three Mile Island plants.

Personnel and equipment access hatches are provided for access to the reactor building. In addition, there are other penetrations for piping and electrical conduits.

Class I components and systems whose design must include consideration of seismic effects are listed in Appendix 5A and include such items as emergency core injection system, reactor building atmosphere cooling and washing system, the spent fuel cooling system and shutdown cooling system, reactor control room and equipment, and the post-incident air filtration system. Some of these items are located totally or partially outside of the reactor building.

The facility includes a cooling water intake and pump structure located at the foot of the intake canal about 400 feet from the reactors.

The bedrock at this site is located approximately 20 feet beneath the present ground surface. The surface overburden consists in the upper layer of approximately 3 to 5 feet of surface fill, followed by the natural soil cover consisting of deposits of thinly laminated organic sandy silts and clays interspersed with marine deposits, and in turn overlying a residual limy soil unit derived from the decomposition of the underlying bedrock. The bedrock consists of biogenic carbonates of Tertiary Age. The uppermost bedrock member is that identified as the Inglis member which is characterized by a cream-colored to an occasionally tan, porous, granular, biogenic limestone and dolomite deposited in a shallow marine environment.

The closest evidence of possible faulting occurs at a distance of 3 miles to the east of the site. Studies of the site show no evidence of existence of subsurface faults.

SOURCES OF STRESSES IN REACTOR BUILDING AND CLASS I COMPONENTS

The reactor building is to be designed for the following loadings: dead load; live loads (including roof loads, pipe penetration reactions, and crane loads); internal pressure due to loss-of-coolant accident of 55 psig; test pressure of 63.3 psig; negative pressure of 2.5 psig; accident temperature of 281° F and operating temperature of 110° F; wind loads corresponding to roof line load of 35 psf; tornado wind loading (300 mph tangential wind velocity, external vacuum of 3 psig, and missile loading); prestressing loads; and seismic loading as described next.

The seismic design of the reactor building is based on the response to a maximum horizontal ground acceleration of 0.05g. Also, the design is to be checked to insure no loss of function for an earthquake based on a maximum horizontal ground acceleration of 0.10g.

The piping, internals, and vessel support design procedure is outlined in answer to Question 9.11 of Amendment No. 2. Therein it is noted that these items will be designed for various loading combinations as listed in Table 1, including the design load, the design earthquake and pipe rupture loads. In addition a discussion of modes of deformation of reactor internals, and the allowable deformations are presented in Table 2.

As noted in Appendix 5A, all Class I structures, components and systems will be designed for primary steady-state stresses combined with the appropriate seismic stresses, and where applicable, in accordance with the appropriate codes. In the case of primary steady-state stress combined with the seismic stress resulting from the maximum earthquake, the response is to be limited so that the function of the component, system, or structure shall not be impaired to prevent a safe and orderly shutdown.

COMMENTS ON ADEQUACY OF DESIGN

Foundations

The applicant has proposed to found the mat foundation for Crystal River Unit 3 on a structural fill composed of crushed limestone. The base of the structural fill is planned to be at about elevation 73 and will extend up to about elevation 80. Quality control of the crushed limestone fill and the 98 percent maximum Modified Density (ASTM Test Designation D1557-66-T) requirement as noted in Amendment 3 will be adequate to assure a structural fill with satisfactory stress-strain properties.

Because the exploratory investigation revealed the presence of both open and filled solution cavities in the limestone bedrock beneath the site, the applicant proposes to undertake consolidation grouting beneath the reactor building to about elevation 30 and beneath other structures to about elevation 60. From the information presented in Amendment 3 on the foundation grouting report on Unit 2 and on the report on the test grouting program for Cystal River Unit 3, it appears that the modified split-spaced hole procedure utilized on Unit No. 2 will be adequate for the foundation of Unit 3. The effectiveness in providing a curtain wall around the area to be grouted is illustrated by Figure 5 of Amendment 3 (excerpts from "Foundation Grouting Report Unit 2") which shows a graph of hole order versus unit grout-take. The graph illustrates that grout-takes approach reasonable limits in the Tertiary and Quaternary holes. It is understood that the grouting specifications for the grouting contract are flexible to the extent that the decision on the hole order at which grouting will be stopped is to be decided by the field engineer. It would be our recommendation in the application of this procedure that the unit grout takes be reduced to 0.5 to 1.0 cubic feet per lineal foot of hole before grouting is stopped, and the applicant has proposed such an approach in Amendment 3. We believe the proposed structural fill and grouting program will be adequate to prevent excessive differential settlement of the reactor buildings and appurtenant structures.

Seismic Design

All structures, components, and systems classified in Class I are to be designed for a design earthquake based on a maximum horizontal ground acceleration of 0.05g. Such items are also to be designed for a maximum earthquake based on a maximum horizontal ground acceleration of 0.10g so as not to impair or prevent a safe and orderly shutdown of the plant. These design levels are in agreement with those proposed by the U.S. Coast and Geodetic Survey (Reference 3) and we concur in these design criteria.

The response spectra to be employed in the design are given in Figure 3 of Appendix 2-I. The response spectrum shown is for 5 percent gravity, the design earthquake. The applicant has stated in Amendment 4 that at periods greater than 1.0 second (not shown on the Figure 3 of Appendix 2-I) the spectra do not fall below the normalized El Centro spectra. The response spectra for use for the maximum earthquake loading condition design will be twice the values of the spectra just described. We concur in the use of the spectra as described.

The vertical component of earthquake excitation will be taken as two-thirds of the horizontal component and will be assumed to occur simultaneously with the horizontal component. We concur in this criterion.

It is noted in Section 6 of Appendix 5A that "The respective vertical and horizontal seismic components at any point on the shell will be added by summing the absolute values of the response (i.e., stress, shear, moment, or deflection) of each contributing frequency due to vertical motion and adding the resultants to the corresponding absolute values of the response of each contributing frequency due to horizontal motion." The seismic stresses are then to be added directly to the dead load, live load, operating loads, and accident (pressure and temperature) loading conditions in accordance with the loading expressions presented in Appendix 5B. The applicant states in Amendment 4 that the seismic stresses are added linearly and directly with the other applicable stresses, and on the basis of this statement, we concur in the design approach.

The damping values to be employed in the dynamic analysis are given in Section 5 of Appendix 5A. These values are to be employed for both the maximum and design earthquake. As noted in answer to Question 9.3 of Amendment 2, a damping value of 5 percent of critical will be used for both the design and maximum earthquake for rocking effects for the foundation. We concur in the use of these values in the design.

The general method of dynamic analysis will be either a modal analysis or will be carried out in accordance with the procedure outlined in Section 6 of Appendix 5A. Further information on the dynamic piping analysis is included in the answer to Question 9.12 of Amendment 2 and provides some clarification to the discussion presented in Section 6 of Appendix 5A. The applicant proposes various boundary conditions as appropriate in amendment 4. It is our recommendation, and the applicant concurs in Amendment 2, that a formal dynamic analysis be performed for Class I structures, equipment and piping, especially for those systems which are vital to plant safety. Additional comments on the analysis of piping system appear later in this report.

The method of analysis to be employed for the reactor building is described in Section 2.2 of Appendix 5C and we concur in the approach as outlined there.

All structures and components classified as Class II are to be designed for a seismic coefficient of 0.05g in accordance with the procedures of the Uniform Building Code. We are in agreement with this approach.

General Design Provisions

The load combination equations to be employed in the design of the reactor building are presented in Section 1.3 of Appendix 5B. We are in general agreement with the combinations to be employed.

The design stress criteria for the reactor building are presented in Appendix 5B and 5C. It is noted therein that the load deformation behavior of the structure is one of elastic, low strain response. The building will be checked for the factored loads and load combinations, compared with the yield strength of the structure, and the load capacity is to be defined as the upper limit of the elastic behavior of the effective load carrying structural materials. The deformation of the structure is to be such that the compressive strain in the steel liner does not exceed 0.005 in./in. nor to cause average tensile strains to exceed that corresponding to the minimum yield stress. (Membrane tension will be limited to $3\sqrt{f'_c}$ and it is noted further that when principal flexural tension exceeds $6\sqrt{f'_c}$

due to thermal gradients through the wall, nonprestressed reinforcing will be added to resist thermal stresses.) It would be our recommendation that no net membrane tension be permitted in the containment shell but since the above tension limits apply to membrane tension combined with flexural tension arising from pressure or thermal effects, as stated in Amendment 4, we concur in the general design provisions noted.

The reinforcing steel to be employed in the plant will consist of either ASTM A-15, A-408, A-431, or A-432. It is noted in Appendix 5B that arc welding for reinforcing splices will not be employed and that Cadweld splices will be used when required. We are in agreement with this approach.

The liner is to be designed so that the critical buckling stress will be greater than the proportional limit of the steel. Present analysis, according to the PSAR, indicates that the basic accident conditions produce a strain of approximately 0.002 in./in. in the liner. The liner is to be analyzed as a flat plate and the liner anchors, which will be vertical angles, are to be spaced horizontally at 18 in. center to center. Further, the applicant proposes a liner anchor design such that the welds connecting the anchors to the liner and the liner-angle anchors will fail before the liner is breached. Generally, we concur in this design approach for the liner so long as this design detail provides an adequate margin of safety against liner rupture, buckling, and long-term service performance.

A discussion of the general design criteria for handling differential settlements and relative motions under seismic response is presented in Appendix 5A and we are in agreement with the general concepts presented there.

The post-tensioning stressing system to be employed will consist of either the SEEE or the BBRV system. In general, the design concepts to be employed in the prestressing are similar to those employed in other plants such as Turkey Point, Oconee, and Three Mile Island. The reactor tendons, which are unbonded, will be protected from corrosion by insertion of a protective coating in the tendons. The steel portions of the plant will be connected electrically to provide protection against stray currents. It is noted in the PSAR that the tendon inspection capability is provided. It is our recommendation that a reasonable inspection program be implemented at the time of the operating license, especially in view of the location of this plant near a salt water environment.

Piping, Reactor Internals, Reactor Vessel, and Vessel Supports

The design approach to be employed for the piping, and reactor internals, which also would include for the most part the design of the engineered safeguard systems, are to be designed for general criteria as outlined in the PSAR, namely in accordance with applicable ASME codes and procedures outlined in AEC Publication TID-7024. A further more detailed discussion of the design approach is presented in answer to Question 9.11 of Amendment 2.

The pssible modes of deformation of reactor internals are summarized in Table 2 of the Answer to Question 9.11 and involve values labeled "allowable" and "no-loss-of-function." It is noted in the discussion preceding the table that the "no-loss-of-function" deformations could cause safety problems, and that the "allowable" deformations are those that are used as design limits. The applicant states in Amendment 4 that these design limits apply to the most severe combination of seismic and other loadings, and that the stress limits apply to all Class I mechanical systems.

The ASA piping code will be used to establish pipe design limits. The approach presented in Amendment 2 is limited in that it relates solely to the margin of safety with regard to stress levels and does not provide information on the margin of safety with regard to permissible strain or deformation. It is our recommendation that a strain limit also be included to assure an adequate margin of safety, and the strain limit be no greater than 20% of the "uniform strain." The "uniform strain" is interpreted to mean the strain at maximum stress on the stress-strain curve for uniaxial tension for the material. On the basis of this approach we concur in this design procedure.

Instrumentation and Controls

The design of the control instrumentation for seismic effects is discussed in answer to Question 9.13 of Amendment 2. Therein it is noted that "the components in the reactor protection system and safeguard actuation system will suffer no loss of function at accelerations of 0.1g horizontal and 0.067g in vertical condition." We do not concur solely in this approach, for an analysis may show that the instrumentation can be subjected to larger accelerations and possible motions such as tilting. However, in Amendment 4 the applicant proposes that the design will reflect maximum seismic loadings, and we concur in this general design approach.

Flooding

Information concerning flooding of the site is presented in Appendix 2C and in answer to Question 9.12 of Amendment 2. The protection provided against flooding appears adequate to us.

Cranes

The polar crane in the reactor building is a Class I component and it is noted in Appendix 5A that the design will be made to insure stability during an earthquake. It is noted in answer to Question 9.10 that other handling bridges which are not considered Class I equipment are also provided with anti-derailing devices. The design criteria for the cranes are acceptable to us.

Penetrations

It is noted in Section 7 of Appendix 5B that the penetrations will be designed for the load combinations applicable to the reactor building and will be analyzed by using the finite element technique developed by the Franklin Institute Research Laboratories. Smaller penetrations will be designed in accordance with published and accepted procedures as noted in the discussion presented in Appendix 5B. We are in general agreement with the design approaches outlined briefly in Section 7 of Appendix 5B.

CONCLUSIONS

In line with the design goal of providing serviceable structures and components with a reserve in strength and ductility, and on the basis of information presented, we believe the design criteria outlined for the containment and other Class I components including the reactor internals, and piping, vessels and supports, can provide an adequate margin of safety for seismic resistance.

N M Newmark

REFERENCES

1. "Preliminary Safety Analysis Report, Vols. 1, 2, 3, and Appendices," Crystal River Units 3 and 4 Nuclear Generating Plant, Florida Power Corporation, 1967.

2. "Preliminary Safety Analysis Report, Amendments 1, 2, 3, 4, and 5," Crystal River Units 3 and 4 Nuclear Generating Plant, Florida Power Corporation, 1968.

3. "Report on the Site Seismicity for the Crystal River Nuclear Plant, Florida," U.S. Coast and Geodetic Survey, Rockville, Maryland, March 13, 1968.

DEPARTMENT OF THE ARMY
COASTAL ENGINEERING RESEARCH CENTER
5201 LITTLE FALLS ROAD, N.W.
WASHINGTON, D.C. 20016

CEREN

26 February 1968

FEB 28 1968

Mr. Roger S. Boyd
Asst. Director for Reactor Project
Division of Reactor Licensing
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Boyd:

Reference is made to your letter of 28 August 1967, and our letters of 16 October 1967 and 22 November 1967, regarding Dockets, Nos.' 50-302 and 50-303, Florida Power Corporation's Crystal River Units 3 and 4 Nuclear Generating Plant and Amendments 1 and 2 of the application. Mr. R. A. Jachowski of the CERC staff has reviewed this application from the viewpoint of the storm surge, design wave height and wave period, wave runup and water level setdown (minimum tide level) as associated with the Probable Maximum Hurricane (PMH) and related to the plant site.

The design water level of 21.4 feet above MLW, based on the applicant's PMH parameters, appears valid although final decision on the acceptance of these assumptions should await the results of a study of the PMH parameters for the Atlantic and Gulf Coasts of the United States currently being prepared by the Hydrometeorological Branch of the U. S. Weather Bureau. The PMH parameter study was requested by this office and approved by the Office, Chief of Engineers (Corps of Engineers) subsequent to our letter of 22 November 1967.

The data submitted in the Amendments regarding the wave runup model studies and the water level setdown analysis provides a reasonable approach to the problem. Based on the design water level of 21.4 feet about MLW, the plant base elevation of 118.5 feet FPC datum (30.5 feet above MLW) provides the necessary height in elevation to prevent significant wave overtopping, but any significant increase in the PMH parameters can affect the design water level, wave runup and overtopping levels.

CEREN
Mr. Roger S. Boyd

26 February 1968

It is therefore our opinion that the applicant (Florida Power Corporation) should be required to re-evaluate the selection of maximum design water level, wave runup, and wave overtopping based on the results of the U. S. Weather Bureau study of the Probable Maximum Hurricane Parameters when they become available, about 1 May 1968.

Sincerely yours,

Joseph M. Caldwell
JOSEPH M. CALDWELL
Acting Director

APPENDIX H-1

UNITED STATES
DEPARTMENT OF THE INTERIOR
FISH AND WILDLIFE SERVICE
WASHINGTON, D. C. 20240

IN REPLY REFER TO:

[illegible] 1 2 [illegible]

Mr. Harold L. Price
Director of Regulations
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Price:

This is in reply to Mr. Boyd's letter of August 17, 1967, requesting our comments on the application of the Florida Power Corporation for a construction permit for the proposed Crystal River Nuclear Generating Plant Units Nos. 3 and 4, Crystal River, Citrus County, Florida, AEC Docket Nos. 50-302 and 50-303.

The project would be located adjacent to Crystal River Unit No. 1 presently in operation and Unit No. 2 which is under construction on the Gulf of Mexico Coast, 7.5 miles northwest of Crystal River. Two pressurized water reactors, designed for a combined ultimate output of 5,088 thermal megawatts and a net electrical output of approximately 1,758 megawatts, would be used as a power source. The heated effluent from the cooling system will be discharged into a shallow water area between spoil from the condenser water discharge channel and spoil from the Cross Florida Barge Canal. This 350-acre area is open to the Gulf of Mexico on the west at Demory Gap. The bottom of this shallow water area is predominantly hard sand and rock.

Several large bars of "coon" oysters occur within the area and small patches of Cuban shoalweed are found in the vicinity of Demory Gap. Although commercial net fishing in the area is limited, crabbing along this section of Gulf of Mexico coastal marsh is extensive. There is a valuable sport fishery for redfish (channel bass) and seatrout in the estuarine area surrounding the project site during winter months.

According to statements of Florida Power Corporation personnel, water in the area of the proposed cooling water intake reaches a seasonal high temperature of 84° to 86° F. during August, and surface temperature reaches 92° F. during calm days. An anticipated 8° to 10° F. gain in temperature will occur as the cooling water passes over the condensers. The gain in heat will remain more or less constant, while the volume of cooling water required will vary according to sea temperature and operational demands. Water requirements for cooling the conventional coal-powered units vary between 285,000 g.p.m. (3,800 c.f.s.) for a single unit to 600,000 g.p.m. ([illegible],000 c.f.s.) for both units. Estimates of the quantity of water required for the operation of each nuclear unit vary between 668,000 g.p.m. (1,500 c.f.s.) and 1,032,000 g.p.m. (2,300 c.f.s.) with a calculated average of 810,000 g.p.m. (1,800 c.f.s.).

The applicant indicates that the release of radioactive wastes would not exceed maximum permissible limits prescribed in Title 10, Part 20, of the Code of Federal Regulations. Although these limits refer to maximum levels of radioactivity that can occur in drinking water for man, without resulting in any known harmful effects, operation within these limits may not always guarantee that fish and wildlife will be protected from adverse effects. If concentrations in the receiving water were the only consideration, maximum permissible limits would be adequate criteria for determining the safe rate of discharge. However, radioisotopes of many elements are concentrated and stored by organisms that require these elements for their normal metabolic activities. Some organisms concentrate and store radioisotopes of elements not normally required but which are chemically similar to elements essential for metabolism. In both cases, the radionuclides are transferred from one organism to another through various levels of the food chain just as are the nonradioactive elements. These transfers may result in further concentration of radionuclides and a wide dispersion from the project area, particularly by migratory fish, mammals, and birds.

In view of the above, we believe that pre-operational and post-operational radiological surveys should be conducted by the applicant and should include studies of the effects of radionuclides on selected organisms indigenous to the project area which require these waste elements or similar elements for their metabolic activities. In addition, the surveys should include studies to determine current velocities and patterns and temperature measurements at various depths, including the surface. These surveys should be planned in cooperation with the appropriate Federal and State agencies. If it is determined from pre-operational surveys that the release of radioactive effluents at levels permitted under the Code of Federal Regulations would result in harmful concentrations of radioactivity in fish and wildlife, plans should be made to reduce the discharge of radioactivity to acceptable levels. Post-operational surveys should be conducted to evaluate the predictions based on the pre-operational surveys and to serve as a basis for reduction of radioactive levels to insure that no unforeseen damage occurs.

In view of the importance of the sport and commercial fisheries of this section of the Gulf of Mexico, it is imperative that every possible effort be made to protect these valuable resources from radioactive contamination. Therefore, it is recommended that the Florida Power Corporation be required to:

1. Cooperate with the Fish and Wildlife Service, the Federal Water Pollution Control Administration, the Florida State Pollution Control Commission, and its member agencies, which include the State Board of Conservation, the Florida Game and Fresh Water Fish Commission, and the State Board of Health, and other interested State agencies in developing plans for radiological surveys.

2. Conduct or arrange for the conduct of pre-operational radiological surveys, including studies of selected organisms indigenous to the area that concentrate and store radioactive isotopes, and of the environment including water and sediment samples; and studies to determine current velocities and patterns and temperature at various depths. These surveys should be conducted by scientists knowledgeable in the fish and wildlife field.

3. Prepare a report of the pre-operational radiological survey, and provide five copies to the Secretary of the Interior for evaluation prior to project operation.

4. Make modifications in project structures and operations to reduce the discharge of radioactive wastes to acceptable level if it is determined in the pre-operational or the post-operational surveys that the release of radioactive effluent permitted under Title 10, Part 20, Code of Federal Regulations, would result in harmful concentrations of radioactivity in fish and wildlife.

5. Conduct post-radiological surveys, similar to those specified in recommendation 2 above, analyze the data, and prepare and submit reports every six months thereafter or until it has been conclusively demonstrated that no significant adverse conditions exist. Submit five copies of these reports to the Secretary of the Interior for distribution to the appropriate State and Federal agencies for evaluation.

We understand it is the Commission's opinion that its regulatory authority over nuclear power plants involves only those hazards associated with radioactive materials. However, we recommend and urge that before the permit is issued, thermal pollution and any other detrimental effects to fish and wildlife which may result from plant construction and operation be called to the applicant's attention. We recommend further that the applicant be requested to discuss this matter with appropriate State conservation officials and the Fish and Wildlife Service and to develop measures to minimize those hazards.

We are unable to determine the acreage of receiving waters which will realize a rise in temperature. Although no damage from the heated effluent from Unit No. 1 is evident at this time, the increased volume of heated water as other units are built and placed into operation may give rise to excessive temperatures in the area between the spoil from the condenser water discharge channel and that of the Cross Florida Barge Canal, particularly in the 350-acre area adjacent to the heated effluent. This latter area varies in depth from 2 to 4 feet and, therefore, contains approximately 1,050 acre-feet of water. Complete displacement of the water in this area could occur every 20 hours with the operation of the conventional coal-powered steam generator presently in use, or every 3.5 hours with the two proposed nuclear steam generators.

Little is known as to effects of thermal pollution in subtropical waters. Shrimp taken from waters exceeding 90° F. are usually flaccid and highly sensitive to stress induced by handling. Large volumes of heated water discharged into an aquatic environment may not only be detrimental to fish life directly, but may also affect these resources indirectly through changes in the ecological community, particularly the food organisms on which fish depend. To measure biological changes in marine organisms and long-term changes in the environment, ecological surveys should be carried out prior to and following plant operation so that comparative data will be available for analysis.

Physical, chemical, and biological aspects of the affected area will need study to achieve an understanding of the impact of the project on fish and wildlife resources and their utilization. Flow studies to determine current and temperature patterns during each season will be needed. The effect of higher water temperatures on the oxygen content will be of concern, particularly during the period from June through September. Determination of changes in the invertebrate fauna will require frequent sampling, probably on a monthly basis for 1 year before the plant is in operation and the first year during operation. Subsequent sampling could be less frequent, as required by ecological conditions.

In view of the Administration's policy to maintain, protect, and improve the quality of our environment, and most particularly the water and air media, we request that the Commission urge the Florida Power Corporation to:

1. Cooperate with the Fish and Wildlife Service, the Federal Water Pollution Control Administration, the Florida State Pollution Control Commission, and its member agencies which include the State Board of Conservation, the Florida

Game and Fresh Water Fish Commission, and the State Board of Health, and other interested State agencies in developing ecological surveys, initiate these studies at least 2 years before reactor operation, and continue them on a regular basis or until it has been conclusively demonstrated that no significant adverse conditions exist.

2. Meet with the above-mentioned Federal and State agencies at frequent intervals to discuss new plans and to evaluate results of existing surveys.

3. Make such modifications in project structures and operation, including but not limited to facilities for cooling discharge waters, as may be determined necessary by the pre-operational or post-operational surveys to protect the fish and wildlife resources of the area.

This opportunity to present our views on the project is appreciated.

Sincerely yours,

C. Orine J. Pantka
Commissioner

APPENDIX H-2

UNITED STATES
DEPARTMENT OF THE INTERIOR
FISH AND WILDLIFE SERVICE
WASHINGTON, D. C. 20240

IN REPLY REFER TO:

APR 1 8 1968

Mr. Harold L. Price
Director of Regulations
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Price:

This is in reply to Mr. Boyd's letter of March 27 requesting our comments on Florida Power Corporation's change in plans to build only one nuclear unit at their Crystal River Site instead of the two units described in the Preliminary Safety Analysis Report.

It is our understanding that the one unit would be designed for an ultimate output of 2,500 thermal megawatts (885 gross electrical) and would require approximately 810,000 gpm (1,800 c.f.s.) of cooling water.

Our comments and recommendations pertaining to the effects of the proposed two-unit plant on fish and wildlife resources, contained in our letter of February 12, are applicable also to the one-unit plant as now proposed. Therefore, we have no additional comments to make on this matter at this time.

The opportunity to express our views is appreciated.

Sincerely yours,

Commissioner