UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION AND ENVIRONMENTAL IMPACT APPRAISAL BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 41 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

Introduction

By letters dated November 29, 1978, January 29, February 28, May 7 and 25, 1979, March 21, April 14 and 30, June 6 and 13, July 22 and 31, and December 5, 1980, and January 29 and July 9, 1981, Florida Power Corporation (FPC or the licensee) requested authorization to operate Crystal River Unit No. 3 (CR-3) at a power level of 2544 MWt. This is an increase from the previously authorized power level of 2452 MWt. This report evaluates the proposed increase in power level and necessary changes in the Technical Specifications (TSs).

I. SAFETY EVALUATION

1.0 Evaluations Completed with Cycle 3 Authorization

Most of the issues involved in the proposed power increase were reviewed as part of our Safety Evaluation for Cycle 3 operation of CR-3 dated August 1, 1980. The following is a list of those issues that we reviewed at the proposed power level of 2544 MWt for Cycle 3 and found acceptable for the proposed power increase:

Fuel Assembly Mechanical Design
Fuel Rod Design, Rod Bow and Internal Pressure
Cladding Collapse
Cladding Stress and Strain
Fuel Operating Experience
Guide Tube Wear
Hold-Down Spring Damage
Cladding Strain and Flow Blockage
Fuel Loading Error
Startup Test Program
Power Distribution
Reactivity TS Changes
Departure from Nucleate Boiling Ratio (DNBR) Parameters
Pressure-Temperature Limits
Thermal Hydraulic TS Changes
Reactor Coolant Pump (RCP) Locked-Rotor Event

2.0 Evaluation of Other Issues

The remaining safety issues related to the proposed power increase are:

Radiological Consequences of Postulated Accidents
Radioactive Waste Management
RCP Power Monitor
Reactor Protection System (RPS) Instrument Inaccuracies.

2.1 Radiological Consequences of Postulated Accidents

2.1.1 Fuel Handling Accident, Control Rod Ejection Accident

We have reviewed the evaluation of the potential radiological consequences of the postulated fuel handling accident in the spent fuel pool and control rod ejection accident in the Safety Evaluation Report (SER) dated July 5, 1974. Both of these accidents were evaluated at a core thermal power level of 2544 MW. Therefore, we conclude that the potential consequences of these accidents are as presented in the July 1974 SER and are within the guidelines of 10 CFR Part 100 and are acceptable.

2.1.2 Main Steam Line Failure, Steam Generator Tube Failure and Waste Gas Decay Tank Failure

The July 5, 1974 SER states that the potential radiological consequences of a steam line failure and steam generator tube failure will be limited to small fractions of the dose guidelines of 10 CFR Part 100 by implementation of TSs limiting the permissible reactor coolant and secondary coolant radioactivity concentrations. The SER also states that doses from a single failure in the gaseous waste system will be limited by TSs on gas decay tank activity. These specifications are in effect at CR-3 and will not be affected by the power increase. Therefore, the conclusions reached in the 1974 SER are not changed by this action.

2.1.3 Loss-of-Coolant Accident (LOCA)

The design basis LOCA was reevaluated based on the licensee's changes to the containment spray system. The changes were evaluated by the NRC staff in the SE dated July 3, 1979, and include the contribution from engineered safety features component leakage outside containment. The evaluation was performed at 2544 MWt and the resulting doses were within the guidelines of 10 CFR Part 100 and, therefore, are acceptable. Therefore, the conclusions reached in the July 3, 1979 SE are unchanged.

2.1.4 Control Room Habitability

The July 5, 1974 SER has evaluated the control building complex ventilation system in both the normal and emergency modes. The SER concluded that the system is acceptable conditioned on resolution of the concern regarding battery room ventilation system. Supplement No. 1 (January 13, 1975) concludes that the concern regarding the battery room ventilation had been resolved.

We have evaluated the potential radiation doses to control room personnel following a LOCA. The evaluation was performed at a core power level of 2544 MWt. This evaluation was based on the licensee's changes to the containment spray system as described by the NRC staff in the SE dated July 3, 1979. The resultant doses are within the guidelines of General Design Criteria 19 contained in Appendix A to 10 CFR Part 50. On this basis, we conclude the design of the control room ventilation system is acceptable.

2.2 Radioactive Waste Management

Our July 5, 1974 SER evaluated the liquid and gaseous waste treatment systems and the solid radioactive waste management system and found them acceptable. The evaluation was performed assuming a thermal power level of 2544 MW. Therefore, the conclusions reached in our SER are not changed.

2.3 Reactor Coolant Pump Power Monitor (RCPPM)

FPC proposed certain modifications to the RPS in conjunction with the request for an increase in licensed power level. These modifications involve installing RCPPM systems on the four RCPs which will trip the reactor on loss of two or more RCPs. As stated in the Accident Analyses of the CR-3 Final Safety Analysis Report (FSAR), in the event that a loss of reactor coolant flow due to failure of one or more of the RCPs was to occur at the present licensed power level of 2452 MWt, the transient is terminated by the present RPS flux-flow trip. However, at thermal power levels greater than 2500 MW, RPS action by the flux-flow comparator is not fast enough in the event of loss of more than one RCP to preclude the minimum DNBR from going below the acceptance criterion of 1.3. Therefore, at power levels above 2500 MWt, nuclear overpower based on RCPPMs must be added to the RPS trip functions as this will reduce the response time of the RPS from 1.4 sec. to .62 sec. and thereby terminate the transient early enough to insure that the minimum DNBR limits are not violated.

The RCPPM is designed to anticipate a loss or reduction of the reactor coolant flow by monitoring RCP power and detecting abnormal power conditions indicative of an inoperable pump. The safety requirements imposed on the CR-3 RCPPMs are detection of loss of power to two or more pumps and a trip output within 240 msec. RCPPM detection of a locked rotor or a sheared shaft are conditions not required by the CR-3 safety analysis but are provided as a standard feature of the RCPPMs. Protection for flow transients involving a locked rotor or sheared shaft is provided by the flux-flow trip. The RCPPM is housed in two sets of cabinets, together with associated bus ducts and cable. One set of cabinets includes current and potential transformers that monitor the three-phase, 6900 volt non-safety electrical power to the RCP motors. A second set of cabinets includes watt transducers that convert the voltage and current measurements to pump power signals that are compared with pump power setpoints in bistables. The bistables drive relays that provide status information (on/off) for each pump to each channel of the existing RPS.

We had determined that the available seismic test data were insufficient to establish qualification to IEEE Std. 344-1975; therefore, the licensee proposed an alternate course. In lieu of demonstrating by type test that seismic events could not prevent proper operation of the transformers, the licensee proposed to install seismically qualified isolation relays between the RCPPMs and the RPS to ensure that the RPS would not be subjected to 6900 volts due to a seismic event. The licensee had previously provided analyses demonstrating that no credible failure including a seismic event could result in failure of the RPS to receive adequate information if in fact a pump motor lost power.

In response to an NRC request to provide a complete analysis verifying that the RPS could not be damaged or disabled due to a fault voltage resulting from any failure or combination of failures of the seismically unqualified equipment or of interconnecting cabling, the licensee determined that the proposed design might not be adequate. The licensee then submitted Reference 10, proposing the addition of optical isolators between the RCPPMs and the RPS and transmitting reports documenting the functional, seismic, and high voltage tests and qualification of the optical isolators.

2.3.1 Setpoints for RCPPM

The licensee provided a revised TS page B 2-7 in Reference 6, additional discussion in Reference 9, and supporting analyses in enclosures to Reference 10. As directed by the NRC in Reference 11 and reflected in the TSs in Reference 12, CR-3 did not increase power for Cycle 3 and hence the need for the RCPPMs and associated TSs did not apply to Cycle 3. In Reference 13, however, the licensee has again requested review with respect to the increased power level.

2.3.2 Evaluation of RCPM's

The RCPPMs serve as sensors for the RPS. Two aspects of the RCPPM application prevent full conformance to IEEE Std. 279 and other normal protection system criteria. First, the measured process variable -- pump power -- is an electrical parameter that is not Class 1E, which dictates that at least a portion of the RCPPM must also be non-class 1E. Second, the need to redundantly monitor the same pump motor power lead for four RPS channels dictates that channel separation and isolation cannot be maintained at the points of measurement.

Conversely, the licensee's design takes advantage of two unusual aspects of the RCPPM application to mitigate these limitations.

First, the safety information required by the RPS is unusually simple: unless normally-open relay contacts are held closed by power from the bistables controlled by the watt transducers, the RPS will interpret the contact openings as loss of pump(s) and will tend to trip. The simplicity of this input is further enhanced by the use of high and low pump power setpoints to provide locked rotor or sheared shaft detection. Although this detection is not required by the plant safety analyses, it is a standard feature of the RCPPM and it does further restrict the possible range of the power signal. Second, instead of using Class 1E power as is customary for RPS sensors, the RCPPMs use the process variable itself -- pump motor power -- as the electrical power source. Thus, the loss of power to a pump motor ensures a loss of power signal from the related RCPPMs.

The licensee has conducted tests to demonstrate the functional capability of the optical isolators to provide the RPS with unambiguous signals indicating the status of the RCPPM output relay contacts. The test results have also demonstrated the ability to fully isolate a 7000 volt signal from a non-current limited source. Finally, the test results have demonstrated the operability of the optical isolators when type-tested to the required environmental conditions and to higher seismic conditions than applied to the remainder of the RCPPM equipment.

Thus, in lieu of seismic qualification of the current and potential transformers and their cabinets, the licensee has demonstrated that in no case including gross failure of the transformer cabinets and contents by an earthquake will the RCPPMs (1) fail to provide loss-of-pump-power information to the RPS when required, or (2) disable any portion of the existing RPS. Hence the effects of not seismically qualifying the transformers and cabinets are limited to the possibility of spurious reactor trips generated during seismic events. Further, the seismic testing performed on sample transformers indicates that the possibility of generating spurious trip signals is small. We find that the RCPPMs including optical isolators are acceptable.

2.3.3 TSs for RCPPMs

Our initial review determined that the RCPPM trip setpoints as identified in TS Table 2.2-1, were not acceptable. Table 2.2-1 as modified in Reference 4 states that the trip setpoint is "Less than three pumps operating". The same statement is also provided in the Allowable (Setpoint) Values column.

Our concern with the TS treatment of setpoints was that the actual pump power setpoints for the RCPPM bistables (between the watt transducer and output relay) were not addressed. The licensee has corrected this omission by proposing modifications to Table 2.2-1 and page B 2-7 of the TSs to require a RCPPM trip if more than one RCP is not operating and to provide the basis for both upper and lower pump power setpoints. The setpoint basis as discussed on page B 2-7 of the TSs (Reference 15) is as follows:

> RCPPMs
>
> In conjunction with the power/imbalance/flow trips, the RCPPMs trip maintains the DNB ratio $\geq$ 1.30 by tripping the reactor due to more than one RCP not operating.
>
> A RCP is considered to be not operating when the power required by the pump motor is $\geq$ 120% or is $\leq$ 70% of the nominal operating power. The nominal operating power decreases from when a pump is first started during heatup and is pumping dense fluid (approximately 7494 Kw) to when a pump is operating at full reactor power and is pumping less dense fluid (approximately 5555 Kw). In order to avoid spurious trips during normal operation, the 120% trip setpoint (9000 Kw) is based on the nominal operating power for a pump during heatup, and the 70% trip setpoint (3900 Kw) is based on the nominal operating power for a pump operating at full reactor power.

Two editorial changes to the proposed TSs submitted by Reference 4 have been discussed with the licensee. In Tables 3.3-1 and 4.3-1, RCPPMs trip should be numbered fourteenth instead of ninth, and the last five trips other than the RCPPM renumbered. This editorial change will avoid renumbering the other five trips in the plant operating procedures. We find the proposed changes to the TS, as modified, acceptable.

2.3.4 References

1. Florida Power Corporation's letter dated November 29, 1978.

2. Florida Power Corporation's letter dated January 29, 1979.

3. Florida Power Corporation's letter dated May 7, 1979.

4. Florida Power Corporation's letter dated May 25, 1979.

5. Florida Power Corporation's letter dated August 14, 1979.

6. Meeting with Florida Power Corporation in Bethesda, Md. on August 24, 1979.

7. Florida Power Corporation's letter dated October 9, 1979.

8. Florida Power Corporation's letter dated February 21, 1980.

9. Crystal River Unit 3 - Cycle 3 Reload Report - BAW-1607 Rev. 1, April 1980.

10. Florida Power Corporations's letter dated December 5, 1980.

11. Letter to Florida Power Corporation from USNRC dated April 4, 1980.

12. License Amendment No. 32 transmitted to Florida Power Corporation by USNRC letter dated August 1, 1980.

13. Florida Power Corporation's letter dated January 29, 1981.

14. Telephone call from R. Bright and W. Klein of Florida Power Corporation to R. Wilson of NRC/ICSB on March 26, 1981.

15. Florida Power Corporation's letter dated July 9, 1981.

2.4 RPS Instrument Inaccuracies

2.4.1 Introduction

By letter dated July 9, 1981, FPC requested that the TSs for CR-3 be altered to reflect the effects of the recently discovered errors in the instrument uncertainty factor and to permit operation at 2544 MWt.

2.4.2 Evaluation

The analyses which support operation of CR-3 at 2544 MWt have been reviewed and approved as part of the reload review for Cycles 2 and 3. Implementation of the power upgrade was delayed pending analyses of the effect of increased "string error" on the RPS and on installation of reactor pump monitor trips. The present request addresses these issues.

The NSSS vendor, Babcock and Wilcox (B&W), recently reviewed the values used for the various instrument uncertainties when establishing RPS setpoints in the 177 fuel assembly plants. They concluded that certain values had been underestimated in the FSAR analyses and investigated the effect of the increased uncertainties on the various trip setpoints. These effects have been included in the proposed TS changes and include reduction of the nuclear overpower and high outlet temperature trips and a narrowing of the flux-flow-imbalance trip envelope. These effects are small (for example the nuclear overpower trip is reduced by 0.6 percent of full power) and are consistent with similar analyses done for other B&W plants. We find the treatment of the "string error" effects acceptable for CR-3.

In summary, we find the proposed resolution of RPS instrument inaccuracies and proposed TS changes to be acceptable. Our conclusion is based on the following:

1. The methods used to obtain the revised specifications are the same as those previously used,

2. The changes due to the revised "string errors" are small and are similar to those for other reactors, and

3. The trends in the changes due to the power increase are correct.

3.0 Conclusion on Safety

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

II. ENVIRONMENTAL IMPACT APPRAISAL

The radiological and nonradiological environmental impacts of operation of CR-3 at a power level of 2544 MWt were evaluated in the Commission's Final Environmental Statement (FES) dated May 1973. The impacts of operation at that power level were found acceptable. No changes to the plant or surrounding environment have occurred since the beginning of plant operation which would affect the conclusions of the FES.

Conclusion and Basis for Negative Declaration

We have concluded that the implementation of the proposed amendment will have no impact on the environment other than that already predicted and described in the FES.

Having reached this conclusion, we have determined that an environmental impact statement need not be prepared for the proposed license amendment and that a Negative Declaration to that effect should be issued.

Dated: July 21, 1981