Florida Power
A Progress Energy Company

Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72

Ref: 10 CFR 50.90

October 31, 2002
3F1002-06

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

Subject: Crystal River Unit 3 – Response to Request for Additional Information Re: Proposed License Amendment Request #270, Revision 0, "Power Uprate to 2568 MWt" (TAC No. MB5289)

References:
1. FPC letter to NRC, dated June 5, 2002, Crystal River Unit 3 – License Amendment Request #270, Revision 0, "Power Uprate to 2568 MWt"
2. FPC letter to NRC, dated September 30, 2002, Crystal River Unit 3 – Response to Request for Additional Information Re: Proposed License Amendment Request #270, Revision 0, "Power Uprate to 2568 MWt" (TAC No. MB5289)

Dear Sir:

During discussions with the NRC staff and through several electronic mail transmittals, the NRC staff has made requests for additional information (RAI) regarding Florida Power Corporation's (FPC) proposed License Amendment Request #270, "Power Uprate to 2568 MWt" (Reference 1), and the previous RAI response (Reference 2). The Attachment to this letter provides the response to these requests.

This letter makes no new regulatory commitments.

The implementation of the power uprate will require setpoint adjustments to the integrated control system, nuclear instrumentation calibrations, and numerous procedure changes. Therefore, Crystal River Unit 3 (CR-3) requests that 90 days be permitted for an implementation period.

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If you have any questions regarding this submittal, please contact Mr. Sid Powell, Supervisor, Licensing and Regulatory Programs at (352) 563-4883.

Sincerely,

Dale E. Young

Dale E. Young
Vice President
Crystal River Nuclear Plant

DEY/pei

Attachment: Response to Request for Additional Information

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

STATE OF FLORIDA

COUNTY OF CITRUS

Dale E. Young states that he is the Vice President, Crystal River Nuclear Plant for Progress Energy; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information, and belief.

Dale E. Young

Dale E. Young
Vice President
Crystal River Nuclear Plant

The foregoing document was acknowledged before me this 31st day of October, 2002, by Dale E. Young.

Janet Schroeder

Signature of Notary Public
State of Florida

JANET SCHROEDER
MY COMMISSION # DD 128063
EXPIRES. June 20, 2006
Bonded Thru Notary Public Underwriters

(Print, type, or stamp Commissioned Name of Notary Public)

Personally Known ✓ -OR- Produced Identification ______

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50 - 302 / LICENSE NUMBER DPR - 72

ATTACHMENT

Response To Request For Additional Information Re: Proposed License Amendment Request #270, Revision 0, Power Uprate to 2568 MWt

Response to Request for Additional Information

Response to Request for Additional Information

NRC Request (per email dated October 10, 2002)

1. Provide a discussion of the instrument setpoint methodology used to determine the effect of power uprate on instrument setpoint and allowable values. If the setpoint methodology has been previously reviewed and approved by the staff then provide the reference to the staff's acceptance of the methodology. Also provide a discussion whether your methodology meets the requirements of 95/95 confidence level and if it does not then provide justification for the level used at CR.

FPC Response

Since no Improved Technical Specification (ITS) setpoints or allowable values are being changed, it was agreed in the conference call between Florida Power Corporation (FPC) and the NRC staff that no response was required for this question.

NRC Request (per email dated October 10, 2002)

2. Your submittal states that calibration and setpoint changes were required for the power uprate. However, no TS changes with the instrument setpoint were submitted with the submittal. This could be based on the fact that no allowable values were changed and there was enough margin between setpoint and allowable values so that TS changes were not required or there was no changes to setpoint for the instrumentation in the TS. If it is based on the fact that there is enough margin then provide the list of instrumentation with both old and new setpoint and allowable values listed for all such instruments. Also provide the copy of a sample calculation for the staff to come to the same conclusion.

FPC Response

With respect to the power level upgrade to 2568 MWt, the only setpoint change is re-spanning certain Integrated Control System modules which control power production functions such as control rod movement, feedwater flow, and turbine control. These inherently have no connection with ITS setpoints and allowable values which are established to limit accident initial conditions and preserve accident analysis assumptions. Therefore, no allowable value changes were requested from the NRC, and no ITS parameter calculations or setpoints are being changed. The nuclear instrumentation will be calibrated to establish 2568 MWt as the new 100% Rated Thermal Power (RTP) level. The existing setpoints will utilize the new RTP but actual setpoints, which are based on percent power, will not change.

NRC Request (per Phone call)

3. Does CR-3 continue to meet GDC 19 after the power uprate?

FPC Response

The Crystal River Unit 3 (CR-3) power uprate does not impact CR-3's compliance with GDC 19. All radiological doses were calculated at 2619 MWt using the alternative source term per 10 CFR 50.67 as permitted in CR-3 License Amendment 199.

NRC Request (Questions concerning Table 6.1 from Attachment C of September 30, 2002 RAI response from phone call on October 15, 2002)

4. In the table header line there is a heading for '% Damping (avg)'...clarify how this average was determined?

FPC Response

The amount of damping the cable stabilizer provides to the stabilized tube system is dependent on the vibration amplitude the tube is experiencing. For small amplitudes of vibration, the damping of the stabilizer will be lower in comparison to a tube experiencing large amplitudes of vibration. The analysis of the damping test data was performed such that the damping associated with the decay of the input signal was determined over a selected number of cycles the stabilized tube system experienced. The number of cycles selected and the range of cycles selected were chosen based upon the vibration amplitude in the response deemed appropriate for fluid-elastic instability. The average damping determined over the selected number of cycles of decay is what is reported in Table 6.1.

NRC Request (Questions concerning Table 6.1 from Attachment C of September 30, 2002 RAI response from phone call on October 15, 2002)

5. In the 'Summary of Damping Results' below the table, the first listing is "Damping of tube (fixed-fixed)." What are the tube boundary conditions for the rest of the damping values presented?

FPC Response

The expression "Fixed-Fixed" refers to the boundary condition at each end of the tube. For this baseline case, the tube is clamped at each end such that there is no clearance between the tube and its supports. This test was relevant to a swollen tube in which the clearance between the tube and tube supports (tubesheet or tube support plates) was reduced. Since a baseline case was desired in which the damping associated with the non-linearity of the tube and tube support interactions was not prevalent, the test "TS-TS" was performed. Therefore, the test "TS-TS" is the same as "Fixed-Fixed".

NRC Request (Questions concerning Table 6.1 from Attachment C of September 30, 2002 RAI response from phone call on October 15, 2002)

6. Items 1 and 9 of Table 6.1 and in the "Summary of Damping Results" below the table, the average damping for a virgin tube in air is higher than that in water, which is contrary to the common belief...please clarify.

FPC Response

As concluded in the second paragraph of Section 6.0 of Attachment "C" of the RAI dated September 30, a gain of approximately 1% damping was observed with a tube pressurized with water and in water. Framatome ANP and FPC agree that the damping of a tube vibrating in water would be more than that vibrating in air. However, the tests as referenced in the 'Summary of Damping Results" were conducted in air with air or water inside the tubes. Thus, in neither case was the tube vibrating in water. When the tube was filled with water, the water moved with the tube and did not contribute to energy dissipation. In addition, the water added mass to the tube. Thus, the damping ratio decreased. This test was performed for information only and concluded that the damping ratio decreased for this condition (not expected to be present in actual OTSG operation). These results were not used to validate or refute any conclusions within the report.

NRC Request (email dated October 3, 2002)

7. RAI 4.18-1 What is the highest expected increase in the flow accelerated corrosion (erosion-corrosion) rates after power uprate?

FPC Response

CR-3 uses the CHECWORKS model to predict the wear rates for the secondary system. A preliminary review using the computer model (PEPSE – Peformance Evaluation of Power System Efficiencies) has shown that the increased flow in the Feedwater System will have the greatest impact on the flow-accelerated corrosion (FAC) rates. Other systems that will experience similar velocity increases are Main Steam and Condensate, but the effect on corrosion rates will not be as great as the Feedwater System due to Main Steam being superheated and Condensate being at a lower temperature. The current model predicts wear in the Feedwater System piping to be less than 0.005 inches per year. Based on Figure 7-8 "Chexal-Horowitz FAC Model, Impact of Liquid Velocity," from the EPRI Guide "Flow-Accelerated Corrosion in Power Plants" (TR-106611, Revision 1), which assumes chemistry and temperature properties are unchanged, the single phase corrosion rate predicts the greatest increase at 300 degrees F. Using this point as a bounding condition, the increase in wear rate for each doubling of the velocity was determined to be approximately 60%. For CR-3, this would mean the Feedwater corrosion rates would increase approximately 0.6 % for the 1% increase in flow velocity. CR-3 has determined that this increase in wear rate is insignificant when compared to the conservatisms inherent in the FAC program. Examples of these are; 1) When calculating remaining life of a component, CR-3 applies a 10% multiplier to the next cycle length, and 2) When inspection data is imported into CHECWORKS during an outage and reanalyzed

(PASS-2), a line correction factor (LCF) is generated which correlates the inspection data to the predicted values. This LCF is then multiplied to the un-inspected components to determine the predicted wear rate. The range for the LCF is 0.5 to 2.5. This means the predicted values from CHECWORKS are adjusted by up to a factor of two to account for actual inspection data.

NRC Request (email dated October 3, 2002)

8. RAI 4.18-2 Describe briefly the flow accelerated corrosion program in your plant.

FPC Response

CR-3 has implemented and maintains a FAC program in accordance with NRC Generic Letter 89-08 and NSAC/202L, "Recommendations for an Effective Flow-Accelerated Corrosion Program." The CR-3 FAC program is an inspection based program which relies on input from the EPRI CHECWORKS computer code, industry operating experience, and engineering judgement to determine inspection scope. Once wear rates have been predicted, the components' time to Code allowable thickness (Tcrit) is calculated. During each outage, components are prioritized and inspected based on the remaining life. Typically, 75 to 125 components are inspected each refueling outage to confirm corrosion rates and component/system acceptability.

NRC Request (phone call October 22, 2002)

9. How was the 2% measurement error accounted for in the rod withdrawl from power accident described on page 9 of 18 of Attachment A of your September 30, 2002 submittal?

Clarification was requested on how it could be concluded that the Rod Withdrawal at Power event is bounded from the standpoint of departure from nucleate boiling ratio (DNBR) and Reactor Coolant System (RCS) overpressure protection at a power level of 2619 MWt (2568 MWt with an applied 2% calorimetric uncertainty).

Calorimetric uncertainty represents the potential for a plant's power measurement instrumentation to deviate from the desired settings. From a safety or accident analysis standpoint, the application of this uncertainty is appropriately imposed as a penalty that would conservatively exacerbate the effects of a transient. In the case of CR-3's Rod Withdrawal at Power, the penalty is applied deterministically to the high flux trip setpoint. Instead of the normal 110% (of 2568 MWt) setpoint, a setpoint of 112% (of 2568 MWt) is used. This method of application is conservative because imposing the 2% penalty on the setpoint allows a longer transient core power excursion. This case maximizes energy deposition to the fuel, clad, and coolant. In contrast, imposing the penalty on initial power non-conservatively minimizes the margin to the high flux trip, limiting the power excursion and consequently, the energy deposition to the aforementioned fission product barriers.

The method's conservatism is further amplified when coupled with the analytical approach for this event. The analytical approach is parametric in nature, that is, the reactivity insertion rate is varied to maximize transient core power, with the accompanying effects on fuel rod temperature rise and the mismatch of primary-to-secondary heat transfer. The mismatch creates an RCS pressure-temperature excursion. These phenomena thus pose a challenge to the two criteria applicable to this event as described in Chapter 14.1.2.3.2 of CR-3's FSAR:

(1) Peak power to remain less than 112%

Limiting peak power ensures that the linear heat rate (LHR) temperature and DNBR limits are not violated. Operating within LHR (kiloWatt/foot) limits, to which core power is an input, ensures that fuel centerline melt does not occur. Similarly, operating within DNBR limits, to which fuel surface heat flux (also based on core power) is an input, minimizes the probability of fuel clad damage. It is recognized that DNBR is also dependent on other factors, such as mass flux, RCS pressure, and temperature. The power level uprate does not result in significant perturbations to these parameters. Therefore, the power level change has the highest impact on DNBR.

(2) Peak RCS pressure to remain below ASME Code pressure limits.

The ASME Boiler and Pressure Vessel Code establishes limits on transient overpressure. RCS overpressure protection is provided by the pressurizer safety valves. The energy transferred to the RCS that must be relieved through the pressurizer safety valves is a direct function of the transient power level.

The near-concurrency of these phenomena result in the situation wherein either the high flux trip or the high RCS pressure trip can occur first. The earlier trip can, therefore, prematurely limit the otherwise adverse effects of the event. As such, the reactivity addition rate parametric study, combined with sensitivity study results for reactor physics parameters, is used to identify the case wherein the high flux trip and high RCS pressure trip coincide. This case has been determined to be limiting because it maximizes both power and RCS pressure, posing the maximum challenge to the acceptance criteria.