8005150051

Jua

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

May 2, 1980

Docket No. 50-302

Mr. J. A. Hancock
Director, Nuclear Operations
Florida Power Corporation
P. O. Box 14042, Mail Stop C-4
St. Petersburg, Florida 33733

Dear Mr. Hancock:

We have reviewed your submittal of March 12, 1980 regarding the February 26, 1980 event at Crystal River Unit No. 3 (CR-3) and have determined that the additional information identified in the enclosure is necessary to continue our review. Please provide this information in a timely manner so that we may prepare our Safety Evaluation that enables restart of CR-3. In order to expedite our review we previously sent you these questions in draft form; the enclosed final form represents only editorial changes from the draft.

Sincerely,

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosure:
Request for Additional
Information

cc w/enclosure:
See next page

cc w/enclosure(s):
Mr. S. A. Brandimore
Vice President and General Counsel
P. O. Box 14042
St. Petersburg, Florida 33733

Mr. Wilbur Langely, Chairman
Board of County Commissioners
Citrus County
Iverness, Florida 36250

U. S. Environmental Protection Agency
Region IV Office
ATTN: EIS COORDINATOR
345 Courtland Street, N.E.
Atlanta, Georgia 30308

Director, Technical Assessment
Division
Office of Radiation Programs
(AW-459)
U. S. Environmental Protection Agency
Crystal Mall #2
Arlington, Virginia 20460

Crystal River Public Library
Crystal River, Florida 32629

Mr. J. Shreve
The Public Counsel
Room 4 Holland Bldg.
Tallahassee, Florida 32304

Administrator
Department of Environmental Regulation
Power Plant Siting Section
State of Florida
2600 Blair Stone Road
Tallahassee, Florida 32301

Attorney General
Department of Legal Affairs
The Capitol
Tallahassee, Florida 32304

Dr. William R. Stratton
Los Alamos Scientific Lab
Box 503
Los Alamos, New Mexico 87544

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
Suite 420, 7735 Old Georgetown Road
Bethesda, Maryland 20014

Bureau of Intergovernmental
Relations
660 Apalachee Parkway
Tallahassee, Florida 32304

Enclosure

CRYSTAL RIVER UNIT 3
REQUEST FOR ADDITIONAL INFORMATION
RESTART AFTER 2/26/80 INCIDENT

Part I: Areas Addressed by Florida Power Corporation

I.1. Your March 12, 1980 response states that the only sure way to determine which indicators or recorders are good (following a power loss) is by observing their operation and response to transient conditions. This situation is unacceptable. Describe those plant modifications to be made to provide positive and obvious indication of the validity of information presented by indicators and recorders for partial, total, a.c. and d.c. power loss situations.

I.2. Your March 12, 1980 response indicates that the control circuitry for the PORV will be modified to include an interlock which will preclude PORV opening on loss of 24 VDC power. Does this interlock depend upon the operation of the NNI Power Monitor? Will the interlock prevent automatic operation of the PORV between the time of a loss of either the positive/negative d.c. power and the time of the tripping of the d.c. supplies (i.e., S1 and S2)? Confirm that this interlock will not interfere with the manual operation of the PORV.

I.3. Your March 12, 1980 response indicates that procedures will be revised, as necessary, to cover methods of recognition of loss of a.c. vital bus power to the NNI/ICS. Confirm that these procedures will cover also the loss of any d.c. bus to the NNI/ICS. Describe the automatic means to alert the operator to a loss of vital a.c. or a NNI/ICS bus. If one considers the loss of such a bus as the "event-initiator" and a "single failure" eliminates the main annunciator panel (as has occurred at other facilities), what automatic means will directly identify the bus failure for the operator?

I.4. Confirm that the training provided for operators and I&C technicians prior to plant restart will include a control room demonstration of the loss of various a.c./d.c. power supplies, confirming the loss of certain instruments and the availability of alternate instruments.

I.5. Your March 12, 1980 response states that redundant NNI indicators will be provided on the main control panel. We understand that, prior to plant restart, these indicators will be "control grade." Confirm that this classification means that no single failure of an active component can cause the loss of both redundant indicators. Provide the design criteria to which this set of instrumentation is being designed, installed, tested, etc.

I.6. Your March 12, 1980 response describes redundant indicators to be provided. Describe the means to be provided to record certain key parameters in the event of a loss of a.c. from either vital bus to the NNI's. Identify which parameters will be recorded.

I.7. (a) Prior to the February 26, 1980 event, did the plant have the design capability and adequate procedures as to what safety-grade instrumentation and equipment should be used in the event of postulated transients and accidents (including loss of power to non-safety grade instrumentation) to follow the course of the event and to maintain a stable hot shutdown condition?

(b) ...cold shutdown?

(c) List the minimum plant variables that must be monitored to safely reach and maintain hot shutdown; ...cold shutdown.

(d) Describe any modifications you would propose that would provide safety-grade instrumentation and procedures to follow the course of transients and maintain hot shutdown; ...cold shutdown.

Part II: Areas Not Yet Addressed by Florida Power Corporation

II.2. The fact that the plant computer overloaded during the February 26, 1980 event with 1400 alarms present is clear evidence that there were too many alarms for the reactor operator to handle. The main alarm panels (including alarm printers, etc.) should provide alarms only at the systems-level. Secondary alarm panels should be provided to alarm causes of system-level alarms and other secondary items. For example, the main alarm might say "diesel generator A not available," with the secondary alarm panel indicating the specific cause of this condition. Identify and describe the plant modifications to be made to reduce the number of main alarms to a managable size for the reactor operator to handle. Provide your schedule and major milestones for the completion of these modifications.

II.3. During the recent incident at Crystal River and other events including the Three Mile Island accident, there has been a major reliance on the core exit thermocouples to provide important core cooling information. These experiences have made clear the need to display core exit temperature information automatically and continuously on the main control board. The values of the 2, 3 or 4 hottest thermocouples in each core quadrant should be continuously indicated in a core-mimic graphic display on the main control board. Identify and describe those plant modifications to be made to automatically display core exit temperature information on the main control board. Provide your schedule and major milestones for the completion of these modifications.

II.4. During the recent Crystal River incident the reactor operator balanced the flows of the high pressure injection (HPI) systems. This action was compromised due to the loss of HPI flow instrumentation. Where preplanned manual safety actions are needed, sufficient instrumentation and controls associated with these actions must remain available in spite of the event-initiating conditions and any postulated single failure. Review all procedures to determine what instrumentation and control needs exist to complete all anticipated manual safety actions. Document and provide this review to the NRC, including a list of all such I&C needs. Identify and describe those plant modifications to be made to assure that the I&C needs associated with anticipated manual safety actions can reliably be met. Provide your schedule and major milestones for the completion of these modifications.

II.5. The design feature of tripping both the positive and negative d.c. power supplies when the voltage from either is slightly reduced causes loss of multiple NNI input signals to the ICS. Determine and evaluate the performance of the ICS when either (a) NNI X, (b) NNI Y, (c) ICS X, or (d) ICS Y d.c. power supplies are tripped. This evaluation should include consideration of the appropriateness of the 22 volt setpoint, the affect on ICS and plant response of deleting this trip action, the

frequency of such trip action, the consequences on plant safety and the merits of tripping the ICS on the occurrence of a d.c. power supply trip. Document and provide this evaluation to the NRC. Identify and describe any modifications that this evaluation indicates are desirable. Provide your schedule and major milestones for the completion of these modifications.

II.7. During the recent incident at Crystal River, both steam generator rupture matrices actuated spuriously leaving each isolated momentarily from both main feedwater and emergency feedwater. No actual rupture existed in either steam generator. The design of the rupture matrix must be reviewed and upgraded in two aspects. First, appropriate input parameters must be selected such that isolation of emergency feedwater to a steam generator does not occur except when a significant rupture has in fact occurred. Second, no single failure should cause the spurious isolation of a steam generator from emergency feedwater. Clarify the operator's responsibilities once the EFW system is turned on, in order to control the steam generators and avoid their unnecessary isolation. Identify and describe those plant modifications to be made to eliminate spurious rupture matrix actuation. Provide your schedule and major milestones for completion of these modifications.

II.8. Describe those provisions that exist or will be installed to protect the pressurizer heaters such that loss of NNI power (or control circuitry) will not result in damage to the heaters if the pressurizer level is very low, i.e., the heaters left in a dry condition.

II.9. Other than the PORV, and spray valve, identify all other NII control circuits (i.e., non-ICS). Describe the actions of each of these control circuits for partial, total loss of a.c. or d.c. power. Evaluate these control systems' actions individually and collectively on plant safety. Identify and describe any modifications you propose to preclude or mitigate single or multiple NNI control system actions. Provide your schedule and major milestones for the completion of these modifications.

II.10. The failure mode values of control signals (e.g., T_h, T_{ave}, RC flow, OTSG S/U level and operate level setpoints, etc.) and their collective effect on the RCS/ESF should be considered. Discuss your plans to address this problem.

II.11. In response to #4 of your March 12, 1980 submittal, the OTSG S/U level, the operate level and the steam pressure fail positions are identified as having no effect on control of plant. During the Crystal River 3 February 26, 1980 event, each one of those fail positions contributed to upset the plant control. Explain the discrepancy.