Florida
Power
CORPORATION
Crystal River Unit 3
Docket No. 50-302

August 15, 1994

3F0894-02

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46, Generic Letter 87-02

References: 1. FPC to NRC letter, 3F0893-12, dated August 27, 1993
2. NRC to FPC letter, 3N0494-07, dated April 12, 1994

Dear Sir:

Reference 1 provided Florida Power Corporation's (FPC's) Plant Specific Procedure for resolution of USI A-46. Walkdown of most accessible equipment in the Reactor Building has been completed in accordance with the procedure. Walkdown of additional equipment outside the reactor building will begin in this month. In Reference 2, the NRC requested clarification and additional commitments in several areas. The Attachment 1 to this letter provides FPC's response to those requests. Attachment 2 to this letter provides a systems level description of the methodology for achieving and maintaining hot shutdown following a safe shutdown earthquake.

Sincerely,

P. M. Beard, Jr.
Senior Vice President
Nuclear Operations

Attachment

PMB:AEF

xc: Regional Administrator, Region II
Senior Resident Inspector

NRR Project Manager
J. N. Burford, FP&L

A075

FLORIDA POWER CORPORATION
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION
ON THE
PLANT SPECIFIC PROCEDURE (PSP)
FOR SEISMIC VERIFICATION OF NUCLEAR PLANT EQUIPMENT

NRC Request 1 - Safe Shutdown Systems/Duration

"In accordance with GL 87-02, the program scope should include the systems and corresponding equipment necessary to ensure that hot shutdown can be achieved and maintained for 72 hours following a Safe Shutdown Earthquake (SSE)."

FPC Response

The Crystal River 3 (CR-3) Technical Specifications define HOT STANDBY as k_{eff} less than 0.99 and T_{ave} greater than 280 °F. HOT SHUTDOWN is defined as k_{eff} less than 0.99 and T_{ave} less than 280 °F and greater than 200 °F. The FPC A-46 program scope includes the systems and corresponding equipment necessary to ensure that HOT STANDBY can be achieved and maintained for 72 hours following an SSE. It is not possible to cool down on natural circulation below about 290 °F in 72 hours because of limited atmospheric steam dump capacity. Achievement of HOT SHUTDOWN following design basis events (such as SSE or Loss of Off-Site Power) is not part of the design or licensing basis for Crystal River 3.

NRC Request 2 - Electrical Relays

"Since the likelihood of encountering an SSE in the range of 0.1g to 0.15g peak ground acceleration during the remaining licensed term of your facility is low, it is unlikely that a potential seismic event would produce vibratory ground motion of sufficient intensity to cause a significant number of relays to experience chatter, especially if it is confirmed that the anchorages for the relays and the equipment housing them are sufficient to withstand a design basis earthquake. For the small number of relays which may experience chatter and cause undesirable effects on safe shutdown equipment, appropriate operator action may be sufficient to cope with the undesirable effects (e.g., reset the relay, work around any affected equipment, etc.) within the time needed to avoid core damage. Thus, a reduced scope of electrical relay evaluation would satisfy the intent of the USI A-46 concern regarding potential seismic-induced relay malfunction subject to the following:

a. Confirmation, by plant walkdowns, that all essential relays in the safe shutdown path are properly installed; i.e., installed per design drawings with adequate anchorages. This may be accomplished by a confirmatory walkdown of a sample population of the safe shutdown relays."

FPC Response

Sections 1.3.2, 1.3.5, and 4.0 of the PSP will be modified to more completely describe the relay verification performed by the seismic capability engineers (SCE's) during the plant walkdown. In addition, a caveat will be added to Appendix B for each of the following equipment classes:

1. Motor Control Centers,

2. Low Voltage Switchgear,
3. Medium Voltage Switchgear,
4. Transformers,
14. Distribution Panels,
16. Battery Chargers and Inverters,
18. Instruments on Racks, and
20. Instrumentation and Control Panels and Cabinets.

The caveat will read:

> "Relays Properly Installed and Supported. All observable relays should be properly installed as intended by the manufacturer. Relays should be installed on structurally sound members. Any unusual relay orientation or mounting detail should be examined to ensure relay does not experience unusually high in-cabinet amplification. Relay mounting should be visually checked to ensure chatter will not occur due to seismic interactions or other phenomena."

The Screening Evaluation Work Sheets (SEWS) for each of these equipment classes will be modified to add a check-off for this caveat.

This revision to the PSP will be submitted to the NRC by September 2, 1994. None of the equipment walked down to date contains any relays. Thus, this change will have no impact on the work already done.

The CR-3 Safe Shutdown Equipment List (SSEL) conservatively includes all (over 130) safety related panels and cabinets. It also includes over 40 non-safety related cabinets that were considered important to smooth and stable plant operation. No attempt has been made to associate specific relays with individual cabinets because of the time it would require. However, because of the very conservative methodology used for adding these components to the SSEL, there is a high probability that these panels and cabinets, along with the equipment in the other categories noted above, contain all of the relays associated with SSEL equipment.

NRC Request 2 - Electrical Relays (continued)

"...a reduced scope of electrical relay evaluation would satisfy the intent of the USI A-46 concern regarding potential seismic induced relay malfunction subject to the following:

b. A commitment to replace all "Bad Actor Relays" (EPRI NP-7148-SL, Appendix E), which are considered susceptible to chatter at very low vibration levels, during maintenance or modification activities that occur for other reasons for the balance of plant life."

FPC Response

Because most of the relays on the "Bad Actors" list are protective or auxiliary relays (used in 4160 volt switchgear) and the remainder are special purpose or not routinely used, it has been possible to identify all of these relays that are associated with equipment on the SSEL. Seventeen relays have been identified. Fifteen of these are HGA auxiliary relays which are used only to generate alarms which would not be maintained. Momentary chatter of these relays during a seismic event is not considered to be a problem since the alarm will clear when

the strong motion stops. The remaining two are CEH relays in protective circuits associated with the emergency diesel generators. These relays will be replaced during the refueling outage currently scheduled for Spring, 1998.

NRC Request 2 - Electrical Relays (continued)

"...a reduced scope of electrical relay evaluation would satisfy the intent of the USI A-46 concern regarding potential seismic induced relay malfunction subject to the following:

c. A commitment to develop a top-level procedure for coping with the consequences of relay chatter. The purpose of this procedure is to ensure that operator action would be sufficient to cope with the malfunction of the "Bad Actor Relays," or any other relays in the safe shutdown path that may potentially chatter. This procedure should alert operators to the potential for seismically-induced relay chattering, describe the expected effects and diagnostic tools available to the operators, and describe the methods for coping with the situation."

FPC Response

Abnormal Procedure AP-961, Earthquake, will be enhanced to respond to this request. Guidance will be added to alert operators to the consequences of mis-operation of the sudden pressure relays which protect the off-site power transformer and the backup engineered safeguards transformer (the normal power sources for the engineered safeguards buses at CR-3). Compensatory measures will be suggested in the procedure.

In addition, the procedure will be revised to provide more generic guidance to assist operators in recognizing and mitigating the effects of contact chatter of other relays not considered to be "Bad Actors." Emphasis will be given to the credible results of contact chatter of normally deenergized relays since these contacts have the highest probability of chattering during a seismic event.

NRC Request 3 - Anchorage

"Section 4.4 of the PSP indicates that the preferred method to determine the adequacy of anchorage is through inspection and judgement of the Seismic Capability Engineers (SCE). This is not acceptable to the staff. As a minimum, Section 4.4 of the GIP, Revision 2, should be thoroughly implemented, not by the SCE's judgement but by the hardware verification using the procedures provided in Appendix C of the GIP, Revision 2."

FPC Response

The anchorage checks made in accordance with Section 4.1.4 of the PSP are done by experienced Structural Engineers. These engineers have completed all training required by the PSP (which is the same as the GIP, Revision 2). Both SCE's participating in the walkdown team must agree that the anchorage is adequate, based on visual assessment, drawing review, or, if required, calculations. These methods, or any other appropriate method, will be used to reach a consensus on anchorage adequacy. The basis for the conclusion of adequacy will be documented and included with the walkdown documentation to allow third party verification. If both SCE's do not agree that the anchorage is adequate, the component will be treated as an outlier and handled in accordance with Section 5 of the PSP. This

process meets the intent of the GIP, Revision 2 in that their combined expertise is considered adequate to determine the degree of rigor needed to evaluate the adequacy of anchorage of equipment at a low seismic risk site such as CR-3. The rationale for this position is described in greater detail in Appendix E of the Technical Basis for the Plant Specific Procedure to Resolve NRC Generic Letter 87-02 (hereafter referred to as the Technical Bases Document). The cost of performing more detailed evaluations, as described in the GIP, is not considered to be a worthwhile or prudent expenditure, since the action does not produce a safety benefit commensurate with the cost.

NRC Request 4 - Equipment Walkdowns/Evaluations

"The PSP indicates that you are implementing the GIP, Revision 2, guidelines for verifying the adequacy of the tanks and heat exchangers. This is acceptable. Special emphasis should be placed on safe shutdown tanks; tanks which do not satisfy the intent of the GIP criteria should be modified to ensure that they do. In addition, with regard to above-ground vertical tanks, if the resolution of USI A-40, "Seismic Design Criteria," is to be achieved through the resolution of USI A-46, then the USI A-46 implementation program must include all tanks in the scope of USI A-40 (i.e., all safety-related, or Category I, above-ground vertical tanks) even if they are not in the safe shutdown path. For the remaining equipment in the USI A-46 scope (e.g., pumps, valves, cabinets, raceways, etc.), perform confirmatory walkdowns and engineering evaluations to demonstrate that the safe shutdown path equipment satisfies the intent of the GL. As warranted, appropriate action should be taken to restore and ensure the functional operability of the equipment, during and following a design basis SSE, in accordance with design requirements."

FPC Response

The guidelines in the PSP for evaluating tanks and heat exchangers are identical to those in the GIP. Any tanks or heat exchangers which do not meet the intent of the PSP criteria will be treated as outliers in accordance with Section 7.5 and Chapter 5 of the PSP. Resolution of outliers will be communicated to the NRC in accordance with Section 9.1 of the PSP.

With regard to resolution of USI A-40, it would appear your request is a change from the previous NRC position as documented in NUREG 1211. Nevertheless, a review of the CR-3 Configuration Management Information System (CMIS) data base was done and it was determined that there were ten safety related, above ground vertical tanks that were not on the SSEL. Two of these tanks are no longer in use and thus require no seismic evaluation. Three of the tanks (the core flood tanks and the pressurizer) are an integral part of the Nuclear Steam Supply System, and are therefore exempt from seismic review. Two of the tanks are mounted on the emergency diesel generator skids. These tanks will be evaluated as part of the diesel generator and are excluded from explicit inclusion on the SSEL by the rule of the box. (See PSP Section 3.3.3.) The remaining three tanks are the waste gas decay tanks. Because the increase in scope is so small, these three tanks will be added to the SSEL. Therefore, the FPC program should be adequate to resolve USI A-40 for CR-3.

The remaining equipment in the USI A-46 scope will be walked down in accordance with the PSP. Outliers will be handled in accordance with Chapter 5. Corrective action will be taken as deemed appropriate. Results will be communicated to the NRC in accordance with PSP Section 9.1.

NRC Request 5 - Cables and Conduit Raceways

"Your conclusion that the raceway systems need not be evaluated by any criteria for the resolution of USI A-46 is not acceptable. However, a reduced scope based on original design, prior re-evaluation, and analytical evaluations with appropriate documentation may be acceptable."

FPC Response

The raceway systems at CR-3 have been evaluated against specific engineering criteria. In 1983, FPC conducted a walkdown of all of the safety related cable trays. For each cable tray support, the geometry was documented and an assessment of the actual hanger loading was made. With this information, each support was seismically analyzed to determine if the allowable loading had been exceeded. Hangers with excessive loading were modified. At the same time, a program was put in place to track the loading on every safety related cable tray hanger so that none would become overloaded in the future.

In addition, as a part of the preparations for resolution of USI A-46, a limited walkdown was performed by two nationally recognized seismic capability engineers and no concerns were identified. Appendix B of the Technical Basis document provides, in detail, the specific criteria and the bases for FPC's conclusion that no additional work needs to be done to verify the seismic adequacy of cables and conduit raceways at CR-3.

FPC's conclusion is also supported by the outstanding performance of literally thousands of raceways in the earthquake sites visited by the teams. These raceways were not designed for earthquakes, but nonetheless experienced earthquakes much larger then the CR-3 SSE. From this outstanding performance record, and considering that the CR-3 raceways were originally designed for earthquakes and later reevaluated for earthquake capacity, it is reasonable to conclude that there is absolutely no risk from earthquake induced cable failures at CR-3. Thus, there is no benefit to offset the cost of performing a more detailed evaluation. It is imprudent to incur a cost that produces no benefit.

Cables and conduit raceways were added to the scope of A-46 during the development of the GIP at the request of some utilities whose raceway design had previously been an issue with the NRC. Their intent was to use the A-46 walkdown as a means to close the issue. The adequacy of the CR-3 raceway design has never been questioned by the NRC. Therefore, notwithstanding the arguments presented above, for CR-3, cables and conduit raceways should be considered beyond the scope of A-46.

NRC Request 6 - Other

"With regard to Appendix D to Attachment 2 and Appendix B to Attachment 3, certain of your proposed deviations from the GIP generic caveats for 23 classes of equipment are not consistent with our positions. Specifically, caveats concerning anchorages, relays, and other related aspects (e.g., attached weight, door, base isolation, etc.) should be revised consistent with our positions described in this letter. Please provide detailed technical bases for deviations from other GIP caveats related to items which are not specifically addressed in the criteria stated above. The technical bases for these deviations should be expanded beyond the argument of low seismicity at the site."

FPC Response

FPC's commitments for anchorage and relay checks are described and justified in responses to other information requests in this document. Other deviations from the caveats in the GIP are based on engineering judgement, the low seismic risk, and the low probability that adverse conditions will result if a seismic event should occur. Simply stated, there is insufficient benefit to justify the cost. Detailed technical justification is unnecessary.

Additional NRC Requests for Information

"(A) Your August 27, 1993, letter indicates the specific design features of your facility are such that the required single failure assumption would require a longer duration than 72 hours to achieve hot shutdown. This is acceptable provided you confirm that adequate water sources and sufficiently redundant features exist to maintain the safe shutdown functions in the event of a single failure during and following an SSE. Also, use of combined water sources from seismically and non-seismically qualified tanks to provide water for maintaining hot shutdown for at least 72 hours, would satisfy the intent of the GL. A reasonable qualitative engineering evaluation would be required to ensure that the "non-seismically" qualified storage tanks and equipment necessary to transfer water from these tanks to the reactor will be functionally operable and available during and following a design basis SSE. You should confirm that vital support systems within seismic and "non-seismic" safe shutdown path would be functional during and following an SSE. Appropriate procedures must be in place to direct plant operators to use the alternative water sources, when necessary, to maintain hot shutdown continuously for at least 72 hours. You should confirm that these actions have been accomplished."

FPC Response

Adequate water sources and sufficiently redundant features exist to maintain the safe shutdown functions in the event of a single active failure during and following an SSE. This includes the water required to support decay heat removal and cooldown. The cooldown to 290°F described in response to NRC Request 1 assumes the capability to steam both steam generators to atmosphere. Failure of either atmospheric dump valve or associated controls would result in a higher temperature at the end of 72 hours.

Three sources of water would be used to maintain the safe shutdown functions for 72 hours without off-site power; the emergency feedwater tank, the condensate storage tank, and the main condenser hotwell. All three sources can be connected to the suction of the emergency feedwater pumps using installed piping and existing procedures. The combination of these sources provides sufficient water to support decay heat removal and cooldown for 72 hours. The emergency feedwater tank and the condensate storage tank will be reviewed for seismic adequacy in accordance with Chapter 7 of the PSP. The main condenser hotwell will be handled as an outlier in accordance with Section 7.5 and Chapter 5 of the PSP.

Additional NRC Requests for Information (continued)

"(B) With reference to your August 27, 1993 letter:

1. *Attachment 2 (Table 1, item 7) indicates that the evaluation of tanks and heat exchangers will be limited to the adequacy of their anchorages. This is not consistent with Section 7.2 of Attachment 3 which indicates that you plan to utilize the guidance of GIP-2 in its entirety. Please clarify."*

FPC Response

It was originally anticipated that the review of tanks and heat exchangers would be limited to a review of anchorage. It was later determined that the GIP would be followed in its entirety. This fact was not reflected in the Technical Bases document. Section 7.2 of the PSP accurately reflects FPC's commitment.

Additional NRC Requests for Information (continued)

"(B) With reference to your August 27, 1993 letter: (Continued)

2. *Appendix B of Attachment 2 explains your position that cable raceways need not undergo a case-by-case review. However, Section 8 of the Plant Specific Procedure (PSP) directs users of the PSP to review Section 8.0 of the GIP. Provide clarification as to how the cable and conduit raceways will be evaluated."*

FPC Response

Seismic Capability Engineers are urged to review Section 8.0 of the GIP to ensure they are generally familiar with the GIP criteria. Thus, while conducting walkdowns of other equipment or simply walking through the plant, they are able to informally assess the condition of the cable and conduit raceways. Should any grossly deficient condition exist, it will be noted by the SCE's and appropriate corrective actions will be taken.

Additional NRC Requests for Information (continued)

"(B) With reference to your August 27, 1993 letter: (Continued)

3. *The discussion following Figure A1 of Appendix A, Attachment 2, indicates that all CR3 SSEL equipment, except the outliers (that exceed the SQUG Reference Spectrum in Figure A1), will meet the GIP capacity/demand screening guidelines and commitment. Three items of equipment located in the auxiliary building elevation 168 feet are identified outliers. This is not consistent with your letter dated October 6, 1993. The floor response spectra provided in your October 6, 1993 letter for two elevations in the reactor building also exceed the SQUG Reference Spectrum at high frequencies. Please clarify this discrepancy, and identify, if any, additional outliers and discuss the technical basis for their resolution."*

FPC Response

When the Technical Basis Document was finalized in June, 1993, the 5% damped floor response spectra for elevations 160 and 180.5 of the Reactor Building did not exist. Subsequent to that, it was determined that some SSEL equipment was located at these elevations and floor response spectra would be required. These spectra were prepared and transmitted to the NRC on October 6, 1993. As stated in the second paragraph of our October 6, 1993 submittal, FPC has determined the floor spectra for elevations 160 ft and 180.5 ft of the Reactor Building are not

bounded by the seismic capacity bounding spectrum based on earthquake experience data from the GIP. All equipment at these elevations and any other equipment not bounded by the seismic capacity bounding spectrum within the scope of USI A-46 will be evaluated as outliers for seismic capacity vs. demand in accordance with Section 5 of the PSP.

Additional NRC Requests for Information (continued)

"(B) With reference to your August 27, 1993 letter: (Continued)

4. *Figure A2 of Appendix A displays limited amplification data collected from three Japanese reactors subjected to 19, 18, and 14 earthquakes, respectively. You provided this data to demonstrate that the amplification factors in the frequency range of 11 to 19 Hz are low such that SSEL, including where the in-structure response spectra (demand) exceeds the SQUG Reference Spectrum (capacity), would be acceptable without an outlier analyses. We do not concur with your position. The data in your submittal is not representative of the earthquake experience in the eastern United States with relatively large high-frequency amplifications; and the SQUG Reference Spectrum already includes consideration of the aspects that you described in your submittal relating to low amplification factors. We recommend that when the equipment responses fall outside the SQUG Reference Spectrum, they should be evaluated as outliers to assure that they can withstand the amplified responses without exceeding specified acceptance criteria."*

FPC Response

Notwithstanding the discussion in Appendix A of the Technical Basis Document, all components not bounded by the seismic capacity bounding spectrum within the scope of USI A-46 will be evaluated as outliers for seismic capacity vs. demand in accordance with Section 5 of the PSP. FPC does believe that the reasoning presented in Appendix A of the Technical Basis Document has technical merit, however, and this reasoning may be used as part of the outlier resolution. The basis for FPC's belief in this technical merit is discussed in detail in the following paragraphs.

The Technical Basis Document went to great length to define structural amplification. The key point is that a technically sound measure of structural amplification is based on an average of several earthquake data records. More data provides a better estimate of amplification. A "rule of thumb" is that about 10 earthquake records are required at any given facility to define amplification with good accuracy. As noted above, the Technical Basis Document uses records from three Japanese reactors subjected to 19, 18, and 14 earthquakes. Thus, these data satisfy this "rule of thumb." We are not aware of any eastern U.S. Reactors that have recorded structural amplification data from 10 different earthquakes that would substantiate the NRC position that "[t]he data in your submittal is not representative of the earthquake experience in the eastern United States with relatively large high-frequency amplifications..."

In addition, while the ground motion from some eastern U.S. earthquakes might have relatively large high-frequency amplitude, this is not the same as eastern U.S. earthquakes causing large amplification of the ground motion in nuclear plant structures as suggested above. The Technical Basis Document discusses

structural amplification of ground motion, not high-frequency ground motion amplitude. The two topics may be related, but they are not the same. Moreover, while the ground motion from some eastern U.S. earthquakes might have large high-frequency amplitudes, we consider it speculative to assume this applies to peninsular Florida.

The SQUG Reference Spectrum is based only on ground motion. While the data base facilities did contain equipment at elevation, in developing the SQUG Reference Spectrum, SQUG did not take credit for amplification of motion with elevation in structures. Thus, we cannot agree with the NRC position that "... the SQUG Reference Spectrum already includes consideration of the aspects that you described in your submittal relating to low amplification factors."

With respect to amplitude of ground motion, however, since the SQUG Reference Spectrum is based on ground motion from some very large earthquakes, it is quite correct to note that the SQUG Reference Spectrum already accounts for some very high ground motion amplitudes at high frequencies. Thus, the SQUG Reference Spectrum already accounts for high-frequency amplitudes that might occur in the ground motion at some eastern U.S. sites, but not necessarily at sites in peninsular Florida.

Additional NRC Requests for Information (continued)

"(B) With reference to your August 27, 1993 letter: (Continued)

5. *In the analytical analysis of the cable and conduit raceways, you have presented a capacity check using a derived static load as low as 1.2xDead Load (1.2xDL). Section 8.3.2 of the GIP, Revision 2, reveals that raceway supports should be classified as outliers if the vertical capacity is less than 3.0xDL, and the ground response spectrum is less than the SQUG bounding spectrum. Please perform outlier evaluations for raceway supports which would not meet the 3.0xDL capacity check.*

FPC Response

The discussion in Appendix B of the Technical Basis Document was intended to demonstrate the conservatism associated with the 3.0xDL check for a low seismic site like CR-3. FPC is not aware of any CR-3 cable and conduit raceway supports that would fail to meet a 3.0xDL check. We simply do not believe that any additional seismic evaluation work for the CR-3 cable and conduit raceways will add any safety benefit since the raceways were designed for earthquake loads from the beginning, were re-evaluated in 1983, and thousands of non-seismically designed raceways have performed well in past earthquakes that were much larger than CR-3's SSE.

METHODOLOGY FOR ACHIEVING AND MAINTAINING HOT SHUTDOWN FOLLOWING A DESIGN BASIS SEISMIC EVENT AT CRYSTAL RIVER 3

GENERAL CRITERIA AND GOVERNING ASSUMPTIONS

The general criteria and governing assumptions for the event are as documented in Section 3.2 of the Florida Power Corporation Plant Specific Procedure (PSP) for Seismic Verification of Nuclear Plant Equipment.

SAFE SHUTDOWN FUNCTIONS

The safe shutdown functions which must be maintained following a Safe Shutdown Earthquake (SSE) are described in Section 3.4 of the PSP. They are:

- Decay Heat Removal
- Reactor Coolant System Pressure Control
- Reactor Coolant System Inventory Control
- Reactivity Control

The primary systems and equipment used to perform these functions are described in the following sections.

DECAY HEAT REMOVAL

Decay heat removal is through the steam generators via natural circulation of the Reactor Coolant System (RCS) given a loss of off-site power. If no loss of off-site power occurs, forced circulation would be available. Since natural circulation is a passive process, there is no active failure in the Reactor Coolant System which could prevent it.

Feedwater to the steam generators is provided by the Emergency Feedwater System and is controlled by the Emergency Feedwater Initiation and Control System. Operation of these systems is unaffected by a loss of off-site power. There is no single active failure which would prevent either of these systems from performing it's safety function.

The primary water source for the Emergency Feedwater System is the emergency feedwater tank which has a capacity of 184,000 gallons, with a minimum of 150,000 gallons required by Technical Specifications. The second water source is the 200,000 gallon condensate storage tank. Condensate storage tank inventory is not controlled by Technical Specifications; however, at least 150,000 gallons of water is normally maintained in the tank. The third source of water for the Emergency Feedwater System is the condenser hotwell. Technical Specifications require a minimum of 150,000 gallons of water be maintained in the hotwell.

If the plant is maintained in HOT STANDBY for the 72 hour period, approximately 350,000 gallons of water will be consumed. If the plant is cooled down approximately 430,000 gallons of water will be consumed in the first 72 hours.

Therefore, the three tanks described above contain sufficient water to support decay heat removal and cooldown for 72 hours.

If the available water sources described above are exhausted, plant procedures direct the operators to cool the reactor coolant system directly using the High Pressure Injection System. Water for this cooling mode is supplied from the borated water storage tank. This tank has a capacity of 420,000 gallons with a minimum of 415,200 gallons required by Technical Specifications. When this water source is exhausted core cooling can be maintained by using the Low Pressure Injection System in combination with the High Pressure Injection System. Suction is taken from the Reactor Building sump and recirculated through the decay heat removal heat exchangers, to the High Pressure Injection System, to the reactor. This cooling mode is not affected by a loss of off-site power. It is also not subject to disabling single active failures and can be maintained indefinitely.

Thus, regardless of the availability of condensate grade water core cooling can be maintained for the duration of the event.

Section 3.2.3 of the PSP requires that the plant be cooled down to the entry point for the Decay Heat Removal System. Cooldown of the Reactor Coolant System is controlled by the decay heat removal rate. Removal of decay heat at a rate faster than it is being produced will cause a cooldown of the RCS. Increasing the steaming rate of the steam generators increases the rate at which decay heat is removed. Immediately following a reactor trip, steaming of the generators is mainly controlled by the main steam safety valves. As the reactor neutron power decreases and fission product decay becomes the predominant source of heat, the main steam safety valves close and the steaming rate is controlled by the atmospheric dump valves (and by the turbine bypass valves if off-site power is available). There is no single active failure which can prevent steaming of the generators either immediately following the reactor trip or in the longer term. There are, however, single failures in the longer term which may limit the rate of steaming of the generators or prevent cooldown beyond a certain point. There is no single active failure which would limit the steaming of the generators to the point where a heat-up of the RCS would occur. Some of these single failures could, however, limit the rate at which the RCS can be cooled down. The single failures can be compensated for by manual actions; although some of these manual actions are not proceduralized.

In addition, given a loss of off-site power, it is not possible to cool down to a hot shutdown condition within 72 hours even without additional failures. An additional 52 hours would be needed to cool down to the Decay Heat System entry point (280°F T_{hot}). This is because of the limited capability to relieve steam through the atmospheric dump valves. Plant procedures direct the operators to cool the plant down at a maximum of 10°F per hour on natural circulation. This rate can be maintained for about 24 hours. At that time both atmospheric dump valves will be at their open limit. At the end of 72 hours (when off-site power is restored), the plant will be at 297°F and the cooldown rate will have declined to 0.4°F per hour. The safe shutdown function is met; however, since decay heat is removed continuously throughout the period.

REACTOR COOLANT SYSTEM PRESSURE CONTROL

Reactor Coolant System pressure is controlled by the pressurizer heaters. The pressurizer heaters can be powered from the emergency diesel generators in the event of a loss of off-site power. There is sufficient redundancy in the

pressurizer heaters and associated power sources that no single active failure can disable a sufficient number of pressurizer heaters to cause a loss of the pressure control function.

REACTOR COOLANT SYSTEM INVENTORY CONTROL

Reactor Coolant System inventory is controlled by the Make-up and Purification System. Coolant is extracted from the RCS through a single letdown line. Redundant coolers cool the water and a block orifice reduces the pressure. The coolant is then returned through a single make-up line. Redundant pumps are available to accomplish this function. Operation of this system is unaffected by the availability of off-site power. The letdown and make-up lines are subject to single active failures of control valves in the lines. Redundancy for the letdown line is provided by the reactor coolant pump seal return lines and coolers. Redundancy for the make-up line is provided by the four high pressure injection lines. The cooling water system for the letdown and seal return coolers is not subject to single active failures and is unaffected by a loss of off-site power.

Thus, there is sufficient redundancy that the inventory control function can be maintained despite a loss of off-site power and a coincident single active failure.

REACTIVITY CONTROL

Reactivity control is maintained through the control rods and boric acid in the reactor coolant. Following a reactor trip, the reactivity is initially controlled by the insertion of the control rods. Later it will likely be necessary to add boric acid to compensate for the decay of xenon and the lower RCS temperature. Boric acid from the Chemical Addition System is available from redundant boric acid storage tanks through redundant boric acid pumps. The boric acid is injected into the Make-up and Purification System through a single line. Operation of the system is unaffected by a loss of off-site power. Except for valves in the line between the Chemical Addition System and the Make-up and Purification System, the system is not subject to disabling single failures. Failure of the valves in the single inter-system tie line can be mitigated by operator action. In addition, water from the borated water storage tank can be used to increase the boron concentration in the Reactor Coolant System. This source is not subject to interruption due to single active failures.

Thus, there is sufficient redundancy that the reactivity control function can be maintained despite a loss of off-site power and a single active failure.

CONCLUSION

The above discussion demonstrates that the four safe shutdown functions described the Section 3.4 of the PSP can be maintained for at least 72 hours following a SSE at Crystal River 3. The maintenance of these functions is not dependent on the availability of off-site power and is not subject to disabling single active failures.