UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION OF FLORIDA POWER CORPORATION'S PLANT-SPECIFIC PROCEDURES FOR SEISMIC VERIFICATION OF CRYSTAL RIVER 3 NUCLEAR PLANT EQUIPMENT

RESPONSE TO GENERIC LETTER 87-02 (USI A-46)

DOCKET NUMBER: 50-302

1.0 INTRODUCTION

In December 1980, the NRC designated "Seismic Qualification of Equipment in Operating Plants" as an unresolved safety issue (USI A-46). The safety issue of concern was that equipment in nuclear plants for which construction permit applications had been docketed before about 1972 had not been reviewed according to the 1980-81 licensing criteria for seismic qualification of equipment [i.e., Regulatory Guide 1.100; IEEE Standard 344-1975, and Section 3.10 of the Standard Review Plan (NUREG-0800, July 1981)]. In Generic Letter (GL) 87-02, which was issued in February 1987 to implement the resolution to USI A-46, the staff concluded that the seismic adequacy of certain equipment in operating nuclear power plants should be demonstrated.

In GL 87-02, Supplement 1, dated May 22, 1992, the staff asked Florida Power Corporation (FPC, the licensee) to commit to use either Generic Implementation Procedure, Revision 2 (GIP-2) or other specific criteria and procedures for resolving USI A-46 at Crystal River Unit 3 (CR-3). On September 4, 1992 (Reference 1), in response to the staff's request, rather than providing either a commitment to comply with the entire GIP-2 or a specific set of criteria and procedures, FPC indicated that it is planning to develop a plant-specific procedure, based on the GIP, for verifying the seismic adequacy of the safe shutdown equipment installed at CR-3. By letter dated November 18, 1992 (Reference 2), the staff informed the licensee that the FPC's response of September 4, 1992, was unacceptable because it did not commit to specific criteria and procedures for resolving USI A-46.

In the submittal dated April 16, 1993 (Reference 3), the licensee provided to the staff some information related to the development of floor (in-structure) response spectra to be used for resolving USI A-46. After reviewing that submittal, the staff sent a request for additional information (RAI, Reference 4) for completing its review. The licensee responded to the RAI in submittals dated September 7 and October 6, 1993 (Reference 5). By letter dated December 16, 1993 (Reference 6), the staff approved the licensee's method used to develop the in-structure response spectra and accepted the resulting in-structure response spectra for the resolution of the licensee's USI A-46 program with the provision that the licensee should verify the equipment and anchorages in accordance with Supplement 1 to GL 87-02.

By an August 27, 1993, submittal (Reference 7), the licensee submitted the "Plant-Specific Procedure for Seismic Verification of Nuclear Plant Equipment for Crystal River 3" (CR-3 PSP) and the "Technical Basis for the CR-3 PSP" to resolve GL 87-02. In a letter dated April 12, 1994 (Reference 8), the staff described general criteria that an implementation program should contain to satisfy the intent of GL 87-02 for facilities located in low seismic regions such as CR-3, and identified specific areas in which FPC's current program appeared deficient when evaluated against these criteria. The primary areas of concern involved the lack of an adequate relay evaluation, unacceptable anchorage evaluation approach, lack of supporting information for cable and conduit raceways, and inadequate justifications for proposed deviations from the GIP caveats for 20 classes of equipment.

In response to Reference 8, FPC sent the staff a letter on August 15, 1994 (Reference 9) addressing some of the issues raised by the staff, and committed to submit a revision to CR-3 PSP. By letter dated September 16, 1994 (Reference 10), the licensee submitted Revision 1 to the CR-3 PSP.

The following evaluation is primarily based on the information in the licensee's submittals dated August 27, 1993 (Reference 7), August 15, 1994 (Reference 9), and September 16, 1994 (Reference 10), but also considers the supporting and clarifying information in References 3 through 8. Specifically, this evaluation discusses only the criteria and procedures used by the licensee, and does not include an evaluation of the findings of the plant-specific walkdown. The staff will determine whether or not the licensee's USI A-46 program conforms to the intent of GL 87-02 when the licensee submits for staff review its final summary report documenting the findings of the plant-specific walkdown. In GL 87-02, the staff essentially requested that affected licensees develop a seismic adequacy verification program containing the following major elements (1) a safe shutdown path ensuring that the plant can be brought to and maintained in a hot shutdown condition for a minimum of 72 hours, (2) the mechanical and electrical equipment associated with the path, (3) the tanks and heat exchangers associated with the path, (4) the cable tray and conduit raceway systems associated with the path, and (5) the essential relays associated with the path.

2.0 DISCUSSION AND EVALUATION

The August 27, 1993, submittal (Reference 7) contained three attachments: Attachment 1 described those instances where FPC was unable to comply, in every case, with the guidance in the GIP and the Plant Specific Procedure related to the assumption of a single-failure due to the unique design features of CR-3; Attachment 2 contained the technical basis for the CR-3 PSP to resolve GL 87-02; and Attachment 3 transmitted Revision 0 of the CR-3 PSP. In developing the CR-3 PSP, FPC met the GIP provisions in some aspects, and in others, it only met the intent of the GIP. The September 16, 1994, submittal (Reference 10) transmitted Revision 1 of the CR-3 PSP, addressing some of the issues raised in the NRC letter of April 12, 1994.

2.1 Adequacy of Criteria and Procedures for Safe Shutdown Path

FPC's A-46 program scope encompasses the systems and corresponding equipment necessary to ensure that hot standby can be achieved and maintained for 72 hours following a safe shutdown earthquake (SSE) event. The licensee took an exception from GL 87-02 because it is not possible to cool down on natural circulation below about 290°F in 72 hours due to the limiting capacity of the existing atmospheric steam dump valve. Furthermore, achievement of hot shutdown following design-basis events (such as SSE or loss of offsite power) is not part of the design or licensing basis for CR-3.

The licensee has identified a primary path and an alternate path to achieve and maintain hot standby following a seismic event. Both paths account for the following plant safety functions: reactivity control; inventory control; and residual heat removal. The plant-specific systems available to perform these safety functions are listed in Figures 1, 4, and 5 of Appendix A to Reference 7. The plant-specific equipment and instrumentation which is absolutely necessary to control and monitor safe shutdown functions are listed in Tables 3-1 and 3-2 of Appendix A to Reference 7, respectively.

Initially, decay heat will be removed by automatic operation of the main steam safety valves until the decay heat rate decreases to the point at which the atmospheric dump valves can be used. The emergency feedwater system will be used to supply water to the steam generators. The three sources of water that would be used to maintain the safe shutdown functions for 72 hours and the number of gallons supplied by each are:

emergency feedwater tank	184,000 gallons
condensate storage tank	200,000 gallons
condenser hotwell	150,000 gallons

All three sources can be connected to the suction of the emergency feedwater pumps using existing piping and existing procedures. The combination of these water sources provides sufficient water to support decay heat removal and cooldown for 72 hours. The emergency feedwater tank and condensate storage tank will be reviewed for seismic adequacy. The main condenser hotwell will be handled as an outlier in accordance with Subsection 7.5 and Section 5 of the CR-3 PSP.

In the unlikely event that all of these cooling sources are unavailable, the operator would implement the primary bleed-and-feed mode of cooling that uses (1) the borated water storage tank, which holds 420,000 gallons, and (2) the high pressure injection system. Should this water source be exhausted, core cooling can be maintained by using the low-pressure-injection system in combination with the high-pressure-injection system. Water 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. The licensee confirmed that components and equipment within this cooling path are seismically qualified according to the original plant design and are determined to be seismically adequate.

The numerous water sources, flow paths, and time available for achieving the operational alignments give reasonable assurance that adequate decay heat removal capability is available for 72-hour hot standby. Symptom-based abnormal and emergency operating procedures for lining up cooling sources are available at the site.

The staff concluded that the licensee's approach to achieve and maintain hot standby for 72 hours following an SSE is acceptable if the licensee confirms that the equipment necessary to assure core decay heat removal for 72 hours in both of the safe shutdown paths is seismically adequate.

2.2 Adequacy of Criteria and Procedures for Electrical Relays

For facilities such as CR-3 located in regions with low seismic hazard, the staff has established, in Reference 8, a general framework of criteria that would satisfy the intent of GL 87-02. The staff position as stated in Reference 8 is as follows:

> 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.
>
> 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.
>
> 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 methods for coping with the situation.

The licensee in Reference 9 has adequately resolved items a & b, and has made a commitment to provide specific procedures at a later date to meet the guidelines for item c. The CR-3 PSP Revision 1 (Reference 10) has only incorporated item a. Therefore, incorporation of items b & c in the CR-3 PSP Revision 1, will be verified during the staff's review of the final summary report on the implementation of the A-46 program at the facilities.

2.3 Adequacy of Criteria and Procedures for Equipment Anchorage

The licensee's original position for anchorage evaluation as described in Revision 0 of CR-3 PSP (Attachments 2 and 3 of Reference 7) was that the preferred method to determine the adequacy of the anchorage is through the inspection and judgment of the seismic capability engineers (SCEs) who were to perform a walkdown of the plant for the A-46 program. The NRC staff considered this position unacceptable (Reference 8). In Reference 9, the licensee continued to maintain that the inspection and judgment of the SCEs are adequate for evaluating the anchorage and that it is not a worthwhile or prudent expenditure of resources to perform evaluations in accordance with the procedures in the GIP. The licensee stated that the anchorage checks are made by experienced structural engineers. It also stated that this process meets the intent of the GIP, Revision 2, in that the engineers' combined expertise is adequate for evaluating the anchorage of equipment at a low seismic hazard site such as CR-3.

The staff disagrees with the licensee. The adequacy of equipment anchorage is the most important issue in the USI A-46 program. The determination of the adequacy of the anchorage requires, in addition to engineering judgment, a certain degree of quantitative analyses and hardware verification. The purpose of the A-46 program is to verify that certain electrical and mechanical equipment is qualified to the plant design basis. The relative seismic hazard at a plant is already reflected by its design-basis earthquake magnitude and the minimum ground motion requirement in the siting rule. The GIP, Revision 2 (p. 4-25), states specifically that the screening approach for verifying the seismic adequacy of equipment anchorage is based on a combination of inspection, analyses, and engineering judgment. Analyses and hardware verification using the procedures in Appendix C of the GIP, Revision 2, are essential parts of the evaluation. Therefore, the staff will verify the licensee's implementation of Section 4.4 of the GIP, Revision 2, and the summary report's documentation of the results of anchorage evaluation and how any outliers are handled.

2.4 Adequacy of Criteria and Procedures for Cable and Conduit Raceways

In Reference 7, the licensee stated that the raceways do not need to be evaluated in accordance with the GIP. In support of its conclusion, the licensee stated that its conclusion was based on the good performance of nonseismically designed raceways during past earthquakes, the fact that CR-3 safety-related raceways were originally designed, and later re-evaluated, for an earthquake, and the fact that CR-3 is located in a low seismic region. On the basis of these arguments, the licensee stated that, in its judgment, the raceway system at CR-3 meets the intent of the GIP.

In Reference 8, the staff stated that it disagreed with the licensee's conclusion. However, the staff indicated a reduced scope based on original design, prior re-evaluation, and analytical evaluations with appropriate documentation may be acceptable. In Reference 9, the licensee responded to the staff's position by reiterating its previous position. The licensee continued to maintain that the cable and conduit raceways at CR-3 should be considered beyond the scope of the A-46 program. The licensee did not submit additional technical information in support of its position other than stating that (1) the raceway systems at CR-3 had been evaluated against specific engineering criteria, (2) in addition, as part of the preparations for resolution of USI A-46, a limited walkdown was performed by two nationally recognized SCEs who had not found any problem, and (3) its conclusion was also supported by the outstanding performance of literally thousands of raceway at sites that had experienced numerous earthquakes.

Such reasons for not adhering to the GIP are similar to the original statement in Reference 7 which the staff rejected. These arguments are too qualitative for the staff to evaluate and to reach a positive conclusion. Lacking specific technical data, the staff maintains that an additional evaluation should be performed to ensure that there are no gross plant-specific design and installation problems. The licensee should submit a report that (1) identifies the cable and conduit raceways examined by the SCEs during its plant-specific walkdown, and (2) summarizes the results of the assessment and the bases for the conclusions reached by the SCEs in verifying their seismic adequacy. The report should also detail the criteria and methodology mentioned in Reference 7.

2.5 Adequacy of Criteria and Procedures for Tanks and Heat Exchangers

The review of tanks and heat exchangers for seismic adequacy contains an evaluation of the stability of tank wall, anchorage, support saddle and legs, and the adequacy of piping flexibility to accommodate motion of the tank. The licensee discussed the methodology in Reference 10. The review guidelines the licensee used are based on EPRI Report NP-5228, Revision 1 (Reference 11), which was endorsed in the GIP. The staff, therefore, finds the proposed guidelines acceptable.

As for the A-40 above-ground tanks, the licensee stated that the core flood tanks are an integral part of the Nuclear Steam Supply system (NSSS), and are, therefore, exempt from seismic review. However, the staff believes that the issue of ability of the vertical tanks to withstand postulated loads has not been covered by any programs other than USI A-40. Therefore, the licensee should include these tanks in the A-46 review. The staff, however, recognizes that the main area of investigation of the A-40 tank review is the tank anchorage and shell wall stability. To satisfy these aspects, the licensee should demonstrate the seismic adequacy of the tank anchorages, and either confirm the shell wall stability or demonstrate that the geometry of these tanks is such that they are not subject to appreciable impulsive loads. The staff will evaluate the licensee's assessment of tanks' seismic adequacy during its review of the A-46 implementation summary report.

2.6 Adequacy of Method Used in Developing the In-Structure Response Spectra

In the submittals dated April 16, 1993 (Reference 3), September 7, 1993 and October 6, 1993 (Reference 5), the licensee proposed an approach for developing the in-structure response spectra. The staff issued a letter dated December 16, 1993 (Reference 6), indicating its approval of the proposed methodology. In the letter, the staff concluded that the ground response spectra and the approach used in developing the in-structure response spectra were in accordance with the licensing commitment and were acceptable for the resolution of USI A-46. The staff's acceptance of the proposed methodology is contingent on the licensee's verification of the seismic adequacy of equipment and anchorages in accordance with the staff positions delineated in Sections 2.3 and 2.7 of this safety evaluation and in Reference 8.

2.7 Adequacy of Criteria and Procedures for Addressing the Generic Caveats for Equipment Classes

As stated in Reference 8, the staff indicated that the licensee's proposed deviations from the GIP generic caveats for 20 classes of equipment are inconsistent with the staff's positions and should be revised accordingly. In Reference 10, the licensee revised only part of the relay issues and maintained other original deviations without adequate justifications. As stated in Reference 8, the technical bases for these deviations should be expanded beyond the argument of low seismicity at the site. GIP generic caveats are those undesirable conditions experienced in previous earthquakes or potential weak links within equipment that affect their capacities in resisting the earthquake motions. Therefore, they are the prerequisites for the application of an experience-based approach in addressing the seismic adequacy of equipment in nuclear power plants. The staff will use GIP generic caveats to evaluate the licensee's assessment of equipment seismic adequacy in its A-46 implementation summary report.

3.0 CONCLUSIONS

On the basis of its review of the information in the licensee's submittals dated August 27, 1993 (Reference 7), August 15, 1994 (Reference 9), and September 16, 1994 (Reference 10), and other supporting and clarifying information in References 3 through 8, the staff finds that, pending resolution of open issues stated in this evaluation of the licensee's plant-specific criteria and procedures for the USI A-46 program, and the review of the final summary report documenting the walkdown results, the licensee's program is, in general, adequate to resolve the primary concern of USI A-46. In a brief summary, the staff finds that:

(1) The licensee's approach to achieve and maintain hot standby for 72 hours following an SSE is acceptable provided that the licensee confirms that

the equipment necessary to assure core decay heat removal for 72 hours in both of the safe shutdown paths are seismically adequate.

(2) The licensee's approach to resolve relay issues is acceptable provided that the licensee revises Section 6 of CR-3 PSP, Revision 1, to include its commitments to the staff positions as delineated in Reference 8.

(3) The licensee's approach to evaluate the adequacy of equipment anchorage is not completely acceptable. The staff will verify the licensee's implementation of Section 4.4 of the GIP, Revision 2, and the summary report's documentation of the results of anchorage evaluation and how any outliers are handled.

(4) The licensees' approach to evaluate cable and conduit raceway is not acceptable. The licensee should submit a report summarizing the results of the cable and conduit raceways assessment to verify its adequacy. The report should also detail the criteria and methodology mentioned in Reference 7.

(5) The proposed guidelines for evaluating the seismic adequacy of tanks and heat exchangers are acceptable when proper documentation is provided. The licensee should include the evaluation of seismic adequacy of core flood tanks in the A-46 program activities for resolving USI A-40.

(6) The ground response spectra and the approach used in developing the in-structure response spectra are acceptable provided that the licensee's verification of the seismic adequacy of equipment and anchorages is in accordance with the staff position delineated in Sections 2.3 and 2.7 of this evaluation and in Reference 8.

(7) The proposed criteria and procedures for addressing the generic caveats for equipment classes are not completely acceptable. The staff will use GIP generic caveats to evaluate the licensee's assessment of equipment seismic adequacy in its A-46 implementation summary report.

The details of these summary items are discussed in Section 2.0 of this report.

Principal Contributors: P. Y. Chen, EMEB/DE
S. Kim, ECGB/DE
K. Desai, SRXB/DSSA

Date: May 2, 1996

REFERENCES

1. Letter dated September 4, 1992, from P. Beard (FPC) to NRC on FPC's response to GL 87-02, Supplement 1.

2. Letter dated November 18, 1992, from H.N. Berkow (NRC) to P. Beard (FPC), on the evaluation of FPC's response to GL 87-02, Supplement 1.

3. Letter (with attachments) from P. Beard (FPC) to NRC, on GL 87-02, Supplement 1, dated April 16, 1993.

4. Memorandum from G. Bagchi to H. Berkow, "Request for Additional Information (RAI) on Development of Floor Response Spectra," dated June 11, 1993 (faxed to FPC).

5. Letters from P. Beard (FPC) to NRC, "Responses to RAI on Development of Floor Response Spectra," dated September 7 and October 6, 1993.

6. Letter from H. Silver (NRC) to P. Beard (FPC), "Evaluation of Methods for Developing Floor Response Spectra for the Resolution of Unresolved Safety Issue (USI) A-46 - Crystal River 3," dated December 16, 1993.

7. Letter from P. Beard (FPC) to NRC Document Control Desk, "Generic Letter 87-02, Supplement 1; Verification of Seismic Adequacy of Equipment in Older Operating Nuclear Plants," dated August 27, 1993.

8. Letter from L. Raghavan (NRC) to P. Beard (FPC), "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46, Generic Letter (GL) 87-02, Crystal River Unit 3 (Tac No. M69440)," dated April 12, 1994.

9. Letter from P. Beard (FPC) to NRC Document Control Desk, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46, Generic Letter (GL) 87-02," dated August 15, 1994.

10. Letter from P. Beard (FPC) to NRC Document Control Desk, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46," Revision 1 to FPC's A-46 Program, dated September 16, 1994.

11. EPRI Report NP-5228, Revision 1, "Seismic Verification of Nuclear Plant Equipment Anchorage," Volume 4: "Guidelines for Tanks and Heat Exchangers," Electric Power Research Institute, Palo Alto, CA, prepared by URS Corporation/John A. Blume & Associates, Engineers, June 1991.

Mr. Percy M. Beard, Jr.
Florida Power Corporation

Crystal River Unit No. 3
Generating Plant

cc:
Mr. Rodney E. Gaddy
Corporate Counsel
Florida Power Corporation
MAC-A5A
P.O. Box 14042
St. Petersburg, Florida 33733

Mr. Bruce J. Hickle, Director
Nuclear Plant Operations (NA2C)
Florida Power Corporation
Crystal River Energy Complex
15760 W. Power Line Street
Crystal River, Florida 34428-6708

Mr. Robert B. Borsum
B&W Nuclear Technologies
1700 Rockville Pike, Suite 525
Rockville, Maryland 20852

Mr. Bill Passetti
Office of Radiation Control
Department of Health and
Rehabilitative Services
1317 Winewood Blvd.
Tallahassee, Florida 32399-0700

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

Mr. Joe Myers, Director
Division of Emergency Preparedness
Department of Community Affairs
2740 Centerview Drive
Tallahassee, Florida 32399-2100

Chairman
Board of County Commissioners
Citrus County
110 North Apopka Avenue
Iverness, Florida 34450-4245

Mr. Larry C. Kelley, Director
Nuclear Operations Site Support
(SA2A)
Florida Power Corporation
Crystal River Energy Complex
15760 W. Power Line Street
Crystal River, Florida 34428-6708

Senior Resident Inspector
Crystal River Unit 3
U.S. Nuclear Regulatory Commission
6745 N. Tallahassee Road
Crystal River, Florida 34428

Mr. Gary Boldt
Vice President - Nuclear Production
Florida Power Corporation
Crystal River Energy Complex
15760 W. Power Line Street
Crystal River, Florida 34428-6708

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street N.W., Suite 2900
Atlanta, Georgia 30323

Mr. Kerry Landis
U.S. Nuclear Regulatory Commission
101 Marietta Street, N.W. Suite 2900
Atlanta, Georgia 30323-0199