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FPL

Florida Power & Light Company, P.O. Box 14000, Juno Beach, FL 33408-0420

L-94-107

MAY 5 1994

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Subject: St. Lucie Unit 1
Docket No. 50-335
Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors
Unresolved Safety Issue (USI) A-46
Generic Letter (GL) 87-02

Re: NRC Letter, Jan A. Norris and L. Raghavan to Florida Power & Light Company, "..General Framework of Criteria Which Would Satisfy the Intent of USI A-46 for Facilities Located in Regions with Low Seismic Hazard", dated February 24, 1994

This letter is provided in response to your letter dated February 24, 1994.

Background

Generic Letter 87-02 was issued on February 19, 1987, subject to the provisions of 10 CFR 50.109, Backfit Rule. On June 2, 1988, at the NRC's White Flint offices, FPL made a cost benefit presentation using the NRC's methodology in NUREG 1211, "Regulatory Analysis for Resolution of Unresolved Safety Issue A-46, Seismic Qualification of Equipment in Operating Plants". This presentation demonstrated that the safety benefit to FPL from resolving GL 87-02 was only on the order of $10,000 for each of its nuclear units. With this very low safety benefit, FPL requested that it be totally exempted from GL 87-02.

The NRC indicated that there was disagreement between Industry and NRC contractor calculated probability numbers for Florida earthquakes, but that they understood FPL's position. Recently, NRC contractors resolved NRC's disagreement with Industry calculated probability numbers for Florida earthquakes, publishing the corrected numbers in NUREG 1488, "Revised Livermore Seismic Hazard Estimates for 69 Nuclear Power Plant Sites East of the Rocky Mountains". The revised NRC numbers are in close agreement with Industry numbers. Therefore, FPL's cost benefit analysis on the GL 87-02 issue, using Industry numbers, continues to demonstrate that

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the costs of this seismic program do not yield an adequate safety payback for FPL nuclear plants.

In a letter dated June 16, 1988, the NRC provided FPL with the option of submitting a "scaled back" program or filing a backfit appeal with NRC management. The offer of a "scaled back" program represented a new staff position on GL-87-02 that was acceptable to FPL. On August 4, 1988, by letter L-88-333, FPL submitted Revision 0 of its "scaled back" program.

On August 4, 1989, after a year, the NRC responded to FPL's "scaled back" program and stated that "the staff further finds that the proposed program would be acceptable..." provided FPL address six additional items and "...include them in the procedure for each plant."

On December 13, 1989, by letter L-89-441, FPL submitted Revision 1 of its "scaled back" program in which it included information for five of the six items. Regarding the sixth item, FPL did not agree that addressing relay chatter, per GL 87-02 methodology, would provide any additional safety benefit to the plant. Nevertheless, on September 15, 1993, by letter L-93-171, FPL submitted a relay review program which provides an excellent method for FPL plants to determine relay outliers. This last item completed the original list of six additional NRC requirements on GL 87-02.

FPL has completed a review of the docketed communications between FPL and the NRC, including the most recent letter from the NRC dated February 24, 1994, which contains a number of new staff positions. FPL has acted in good faith and provided the NRC with a comprehensive program for resolving GL 87-02. This program is responsive and reasonable for the low probability numbers identified in NUREG 1488 and FPL's earlier calculations. Any further resources expended by FPL on this issue would not result in improved safety at our nuclear units.

Notwithstanding, FPL is responding to Items 1, 2 and 3 in the NRC's letter of February 24, 1994. These responses, provided in Attachment A, provide additional technical information and explanations of FPL's approach for resolving remaining issues. Additional detail and description on FPL's proposed relay evaluation program is provided as Attachment B.

FPL has also completed the response to the NRC's Request for Additional Information dated June 23, 1993. Our response is provided as Attachment C to this letter.

Summary

FPL has completed the walkdowns of St. Lucie Unit 1 and Turkey Point Units 3 and 4 and a total of 38 outlier items were identified. Of these items, 36 have been resolved, many of them with physical changes to plant systems and equipment. FPL's remaining work on this issue is summarized as follows:

1) The 2 remaining outlier items on Turkey Point will be resolved by September 30, 1994.

2) FPL agrees to make changes to the final reports as described in Attachment C responses to RAI items 4 and 15.

3) Contingent upon NRC's acceptance of our "scaled back" program, FPL agrees to make procedure modifications as described in our response to NRC's criteria 2(c) in Attachment A.

4) Contingent upon NRC's acceptance of our "scaled back" program, FPL agrees to perform a relay evaluation as described in Attachment B.

FPL believes that further NRC requests for work, evaluations, or plant changes would provide no additional safety benefit to our nuclear facilities. FPL requests that the NRC complete the review of our final reports and issue Safety Evaluation Reports for St. Lucie Unit 1 and Turkey Point Units 3 and 4.

Sincerely,

D. N. Paduano Jr
for W. H. Bohlke
Vice President
Nuclear Engineering and Licensing

cc: Stewart D. Ebneter, Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, Turkey Point Plant
Senior Resident Inspector, USNRC, St. Lucie Plant

ATTACHMENT A

ADDITIONAL INFORMATION IN RESPONSE TO NRC LETTER ON GENERIC LETTER 87-02 DATED FEBRUARY 24, 1994

<u>Response to Item 1 of NRC Letter Dated February 24, 1994... Regarding Safe Shutdown Systems/Duration</u>

FPL's program scope includes the systems necessary to ensure that hot shutdown can be achieved and maintained for 8 hours. In our letter of September 15, 1993, we provided information on additional on-site water sources that would allow the units to maintain hot shutdown for 72 hours. The equipment associated with these water sources will not be seismically evaluated or added to FPL's Equipment List. However, based on recent conversations with the staff, we better understand NRC's desire of providing "reasonable assurance" of additional water availability following a postulated seismic event. Therefore, a more detailed description of an operator's procedural actions in lining up cooling sources following a postulated loss of steam generator heat removal is provided below. Please note that FPL and other nuclear utilities have revised emergency operating procedural philosophies following the accident at Three Mile Island (TMI) to "symptom-based" as opposed to "event-based". Reference to this post-TMI requirement and the attached description will be made later in this response.

The following information is supplied to provide a better understanding of the diverse water sources that are available for removing reactor core decay heat at St. Lucie Unit 1 and Turkey Point Units 3 and 4. These sources and operator actions are already a part of site symptom based normal and emergency operating procedures. No procedure changes are required for the lineup and delivery of any of the water sources or off normal cooling modes that are discussed below.

<u>Turkey Point Plant Decay Heat Removal Capabilities:</u>

Following a reactor trip, core decay heat removal would continue via the steam generators and the normal balance of plant systems. While many of these systems and components are not designed and installed to Class- IE criteria, they are generally superior in ruggedness to normal industrial installations. Core decay heat cooling in this mode can continue indefinitely.

In the event that a loss of off-site power occurs coincident with or caused by a postulated seismic event which disrupts normal feedwater supply, site operators would move from normal procedures to the Emergency Operating Procedure (EOP) "Response to Loss of Secondary Heat Sink." The first core decay heat removal cooling path selected by the operator is the Auxiliary Feedwater System (AFW) taking suction from the

Condensate Storage Tank (CST). This alignment was a part of the Seismic Review Team walkdown for Generic Letter (GL)87-02.

While the Technical Specification minimum level for the CST is 180,000 gallons, the CST is normally at 240,000 gallons. Assuming the unit has undergone a full power reactor trip, a 240,000 gallon supply will provide up to 30 hours of core decay heat cooling. This period of time is more than sufficient to conduct post trip surveys and conduct contingency planning if required. Other operational cooling configurations that are available via this procedure are reestablishing main feed water flow, establishing feed flow using the Standby Steam Generator Feed Pumps, and establishing feed flow from Unit 2 (fossil unit immediately to the north of the nuclear plants) or the adjacent nuclear unit assuming it has not lost power. These are all long term cooling sources that can provide indefinite decay heat removal.

Assuming that these cooling sources are unavailable, the operator will move to lining up the numerous other water sources located on-site. These include:

Raw Water Storage Tank #1	500,000 gallons
Raw Water Storage Tank #2	750,000 gallons
Demineralized Water Storage Tank	500,000 gallons

Following 30 hours of cooling from the CST, an additional 190,000 gallons of cooling water is needed in order to provide core decay heat removal for 72 hours. Therefore, any of these three separate water sources has sufficient inventory to provide the additional core cooling time to reach 72 hours of cooling.

In the unlikely event that all of these cooling sources are unavailable the operator would implement the bleed and feed mode of cooling which uses the Refueling Water Storage Tank and the Safety Injection Pumps. This emergency mode of cooling can provide indefinite decay heat removal. Components and equipment within this cooling path are either seismically qualified per original plant design or they were walked down by the GL 87-02 Seismic Review Team and determined to be seismically adequate.

<u>St. Lucie Unit 1 Decay Heat Removal Capabilities:</u>

Following a reactor trip, core decay heat removal would continue via the steam generators and the normal balance of plant systems. While many of these systems and components are not designed and installed to Class IE criteria, they are generally superior in ruggedness to normal industrial installations. Core decay heat cooling in this mode can continue indefinitely.

In the event that a loss of off-site power occurs coincident with or caused by a postulated seismic event which disrupts normal feedwater supply, site operators would move from normal procedures to the Off-Normal Operating Procedure "Auxiliary Feedwater Off-Normal Operating Procedure St. Lucie Unit 1". The first core decay heat removal cooling path selected by the operator is the AFW taking suction from the CST. This alignment was a part of the Seismic Review Team walkdown for GL 87-02.

While the Technical Specification minimum level for the CST is 116,000 gallons, the CST is normally at 240,000 gallons. Assuming the unit has undergone a full power reactor trip, a 240,000 gallon supply will provide up to 19 hours of core decay heat cooling. This period of time is more than sufficient to conduct post trip surveys and conduct contingency planning if required. The next operational cooling configuration available via procedure is aligning the Unit 2 CST to the Unit 1 Auxiliary Feedwater Pumps. This mode of cooling will provide an additional 31 hours of decay heat removal capability from a seismically installed water source.

Assuming that the Unit 2 CST is unavailable, the operator will move to lining up the numerous other water sources located on-site. These include:

City Water Tank #1	500,000 gallons
City Water Tank #2	500,000 gallons
Treated Water Tank	500,000 gallons

Following 19 hours of cooling from the CST, an additional 325,000 gallons of cooling water is needed in order to provide core decay heat removal for 72 hours. Therefore, any of these three separate water sources has sufficient inventory to provide the additional core cooling time to reach 72 hours of cooling for St. Lucie Unit 1.

In the unlikely event that all of these cooling sources are unavailable, the operator would implement the feed and bleed mode of cooling which uses the Refueling Water Storage Tank and the Safety Injection Pumps. This emergency mode of cooling can provide indefinite decay heat removal. Components and equipment within this cooling path are either seismically qualified per original plant design or they were walked down by the GL 87-02 Seismic Review Team and determined to be seismically adequate.

<u>Conclusions on Site Water Source Availability</u>

The above descriptions of the numerous water sources, flow paths and time available for achieving the operational alignments provides reasonable assurance that adequate decay

heat removal capability would be available in the event of a seismic event.

Response to Item 2 of NRC Letter Dated February 24, 1994... Regarding Electrical Relays

FPL has reviewed the three requested actions regarding relays in your letter dated February 24, 1994 and determined that because of the extremely low probability of a seismic event that could cause improper relay action, and the fact that sufficient symptom-based operating procedures for resetting relays are already in place at both sites, the three actions requested would not result in any additional safety benefits at our nuclear stations. We have estimated that the per nuclear unit cost of item (2)a is in the range of $50,000 and that items (2)b and (2)c are in the range of $500,000 when considering:

- The engineering costs of determining the location and location specific functions of all essential relays in the EPRI NP 7148SL Appendix E list (EPRI list),

- The cost of design changes for replacing all relays in the EPRI list,

- The FPL salvage costs for removing all 23 relay types on the EPRI list from our warehouses and the cost of purchasing replacements.

FPL's September 15, 1993 submittal provided the approach for addressing the seismic capabilities of relays for FPL's low seismic sites. This approach provides reasonable assurance that FPL site specific relays on the EPRI list in our A-46 equipment shutdown path will be identified and resolved. The FPL relay evaluation process was provided as a flow chart in our September 15, 1993 submittal. This approach begins with the EPRI list which was compiled from earthquake experience data, tests, Licensee Event Reports, and Information Notices that often represent sites with seismic levels higher than those possible at our low seismic sites.

Many of the relays on the EPRI list have been qualified to IEEE 344-1975 requirements for our seismic levels either on St. Lucie Unit 2 or as replacement equipment or modifications on St. Lucie Unit 1 or Turkey Point Units 3 and 4. Using this information, the EPRI list is a considerably smaller subset for FPL's low seismic probability and low seismic demand.

Included in Attachment B to this letter is more discussion on how FPL's relay program would be conducted along with a modified process flow chart with process steps indicated.

Response to Item (2)a:

Given the extremely low probability of a seismic event at FPL's nuclear sites, FPL does not see any benefit to walking down "...all essential relays in the safe shutdown path..." in order to verify that the relays have been mounted properly. Further, FPL has no reason to believe there are anchorage problems on relays that were installed during original construction, by vendors supplying complete or subassemblies of electrical equipment, by plant personnel for design changes, or by plant maintenance.

Response to Item (2)b:

Generic Letter 87-02 (GL 87-02) was divided into two parts. The first part requested the identification of "outliers" and had a supporting generic Backfit Analysis. Details of FPL's program for identifying relay outliers is provided as Attachment B. The second part of GL 87-02, resolution of "outliers", requires the NRC to justify any resolutions not volunteered by a utility by providing the utility with a plant specific Backfit Analysis. Paragraph (2)b requests FPL to commit to wholesale replacement of all relays on the EPRI list without a plant specific backfit analysis. FPL can not commit to such a program since it is clear that many of the relays on the EPRI list are not "outliers" at FPL nuclear sites. FPL agrees to perform a relay evaluation program per Attachment B. Relays that are determined to be "outliers" by this program will be replaced provided the relay can not be accommodated by plant procedure and/or operational actions.

Response to Item (2)c:

Following the accident at TMI, FPL and other nuclear utilities revised emergency operating procedures to implement a "symptom based" operational philosophy as opposed to "event" based. Existing off-normal procedures at both sites look at loss of function and provide the methods for reestablishing the function. One of these methods is the resetting of relays and breakers. Hence, an event specific procedure for hypothetical anomalous relay action is not necessary and would introduce pre-TMI event based philosophies. This is not in the best interest of current operational philosophies and FPL does not commit to write an event based procedure as requested in Item (2)c.

However, FPL agrees to modify the Turkey Point and St. Lucie site Emergency Plan Implementing procedure "Natural Emergencies" to include a statement to alert operators to the fact that if a seismic event is believed to have occurred and a loss of plant function has followed, unplanned relay actions may be the cause of the loss in function.

Response to Item 3 of NRC Letter Dated February 24, 1994... Regarding Equipment Walkdowns/Evaluations

The docketed Final Reports for Turkey Point Units 3 & 4 and St. Lucie Unit 1 describe confirmatory walkdowns and engineering evaluations to demonstrate that the equipment on FPL's equipment lists satisfy the intent of the Generic Letter. The selection of equipment categories on the FPL lists was based on a maximum return of safety benefit per walkdown dollar for a low seismic utility. Revision of the FPL equipment lists can not be made as the addition of other categories of equipment would not yield any additional safety benefit for the plants. Functional operability of equipment was addressed as part of the walkdown, and was based on the unit experiencing a design basis Safe Shutdown Earthquake (SSE). Additionally, safe shutdown tanks were included in FPL's equipment lists and the USI A-40 tank evaluations for these tanks are complete.

Response to NRC Comments (A) and (B) of NRC Letter Dated February 24, 1994

FPL's response to items (A) and (B) were covered by the above responses to Items 2(a), 2(b), and 2(c).

Response to NRC Comment (C) of NRC Letter Dated February 24, 1994

FPL action on Generic Letter 87-02 received independent review by the NRC outside of the Seismic Qualification Utility Group (SQUG) implementation process. Once the equipment lists and walkdown procedures were complete and the only outstanding issue between FPL and the NRC concerning program acceptability was relays, FPL scheduled and implemented the walkdowns. This schedule placed FPL ahead of the industry in achieving compliance with the intent of Generic Letter 87-02 and USI A-46. In order to assure that our walkdowns were conducted to proper criteria, FPL selected the most senior and experienced walk down team available in the industry. These team members are discussed in our final reports and are either authors or reviewers of the final industry Generic Implementation Procedure (GIP) walkdown criteria that were used by the SQUG. FPL is confident that the Turkey Point and St. Lucie Unit 1 walkdowns were conducted by senior experts very knowledgeable with the criteria and caveats of the GIP. Additional information on this subject is provided below.

Additional Justification on why Valves, Cable Trays and Conduit Systems Were Not in the Walkdown Scope and Additional Information On The Walkdown Review Criteria

Valves, Cable Trays and Conduit Systems

FPL developed a cost effective program for addressing USI A-46 by including nuclear plant components that have experienced

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Valves, Cable Trays and Conduit Systems

FPL developed a cost effective program for addressing USI A-46 by including nuclear plant components that have experienced

the greatest amount of damage in real earthquakes. This philosophy of "equipment selection for maximum payback in plant safety" was discussed with the NRC on June 2, 1988. At that time the NRC staff reviewers were primarily concerned about anchorages on electrical cabinets and large active plant components.

Table 2-4 of Revision 1 to Electric Power Research Institute report NP-6041-SL, "Methodology for Assessment of Nuclear Power Plant Seismic Margin", dated August 1991 contains seismic margins screening criteria for various categories of equipment and subsystems. FPL plants are in the lowest category in that table where the peak spectral acceleration of the 5% damped ground response spectra is below 0.8g. The table indicates that valves, cable tray, and conduit systems are screened at this earthquake level without further evaluation. The only caution is in regard to valves when extremely large operators are attached to valves in 2 inch and under piping. The walkdowns covered a significant portion of the shutdown piping, and this condition was not observed.

FPL concluded that because of the inherent seismic ruggedness of valves, cable trays and conduit systems that walkdown of this equipment would not provide any enhancements to safety at our nuclear plants.

Additional Information On The Walkdown Review Criteria

It is the opinion of the FPL Seismic Review Team (SRT) that all equipment items that were screened during the walkdowns at the FPL plants would also meet the screening criteria in the Rev. 2 Generic Implementation Procedure (GIP) as Supplemented by the NRC SER. This is because the essential inspection requirements and criteria for a given equipment item are the same in both programs. There are four criteria that need to be met:

1. Capacity Vs. Demand
2. Inclusion in the Data Base and Caveat compliance,
3. Anchorage Adequacy, and
4. Seismic Interactions.

A discussion of these four criteria follows:

SEISMIC CAPACITY VS. DEMAND

Seismic Capacity vs. Demand in the FPL program was evaluated at the plant level. It was judged that since the design basis earthquake levels at FPL were significantly lower than the Senior Seismic Review Analysis Panel (SSRAP) Bounding Spectra it was not necessary to compare Capacity vs. Demand for each equipment item as done in the GIP Rev. 2 procedure. Other

facts considered in this conclusion were:

1) the equipment was located at relatively low elevations in the plant,

2) the equipment was located in typical nuclear structures,

3) the amplified response of all equipment at the FPL plants to a design basis event is significantly less that what the equipment in the data base plants experienced during real strong motion earthquakes.

INCLUSION IN THE DATA BASE AND CAVEAT COMPLIANCE

The equipment included in the walkdown at the FPL plants are similar in type and configuration to the equipment at the other A-46 plants and the industry earthquake experience data base. This was confirmed during the walkdown by the expert SRT, and a subsequent review (size and weight) of all walkdown data sheets by Stevenson & Associates.

The best reference available to use as walkdown guidance at the time of the FPL walkdowns was the draft SSRAP Report, Revision 4. At that time, all major issues were well defined with regard to seismic adequacy of equipment and equipment performance during earthquakes. These major issues were included in the Rev. 4 of the SSRAP report. The SRT was well aware of these issues having either authored, developed or reviewed the documents available on the subject of earthquake experience and seismic walkdowns. It is believed that the SRT because of their in-depth knowledge of earthquake experience has superior skills at seeking out any detail that was seismically vulnerable at the FPL plants. The equipment and caveat descriptions in Rev. 4 of the SSRAP report augmented by the SRT knowledge is acceptable for addressing equipment construction adequacy or equipment categories and specific caveats (referred to in the Rev. 2 GIP).

The Rev. 2 GIP was very prescriptive in regard to equipment descriptions and caveats. The approach used during the walkdown at FPL was not prescriptive because of the level of expertise of the SRT. The GIP Rev. 2 procedure can be applied by two engineers with Bachelors' Degrees, a combined 10 years of engineering experience, with one Professional Engineer. The FPL procedure was applied by three engineers, recognized seismic expert, authors of the SSRAP report, the GIP and the EPRI NO-6021 Seismic Margins report, combined 80 years' experience, all with engineering doctoral degrees, and all Professional Engineers.

ANCHORAGE ADEQUACY

The walkdown and supporting calculations were performed prior to the publication of Volume 1 of the EPRI NP-5228, Revision 1 (which was the publication referenced in the Rev. 2 GIP). However, the SRT was aware of changes to be published in the Rev. 1 document due to their unique knowledge of these publications.

The walkdown procedure that was developed by FPL was non-intrusive and did not include bolt tightness checking.

SEISMIC INTERACTIONS

The evaluation for seismic interactions in the FPL program and the GIP Rev. 2 are essentially identical. The FPL walkdowns used the SSRAP report and the EPRI NP 6041 report for guidance.

ATTACHMENT B
ADDITIONAL DESCRIPTION OF FPL RELAY EVALUATION PROCESS IN RESPONSE TO NRC LETTER DATED FEBRUARY 24, 1994

Additional Description of FPL A-46 Relay Evaluation Process

Please note that this description is provided for better understanding of the process. All data at this point are preliminary. Note that the Step numbers correspond to the flow chart attached to this description.

Step 1. Begin with EPRI NP 7148SL Appendix E list of relays (EPRI list).

1.a A broad review of our Total Equipment Data Base (TEDB) for use of the relay models was used. Of the 23 types listed, only 13 are used in <u>any</u> applications (safety, non-safety, safe shutdown at FPL plants).

1.b By taking a closer look at the specific relay type, 2 more types are eliminated. For example, many HFA relays are installed on our units but there are no HFA65D style relays, which is a special spring-balanced timing version of the standard HFA relay. The standard HFA is used only as an auxiliary relay, generally for multiplying contacts. Sudden pressure switches are used in large transformers such as the main, auxiliary and start-up to detect internal transformer faults, not in the A-46 shutdown path.

Thus at the conclusion of Step 1, the EPRI list for FPL is reduced to 11 relay types for further investigation.

Step 2. The next step is to determine if any of these remaining relay types has been qualified to IEEE 344-1975 criteria for our site specific demand. If so, they are sufficiently rugged for our low seismic zone.

2.a For example, GE CFD, GE CFVB, GE IJD, GE PVD, <u>W</u> COM5, <u>W</u> SG are used in equipment on St. Lucie Unit 2 which was specified to meet IEEE 344-1975.

2.b A replacement HGA model 12HGA111J2 was recently ordered for Turkey Point and qualified to IEEE 344-1975.

2.c During the Emergency Power Upgrade project at Turkey Point, the new diesel generators and control panels were specified to IEEE-344-1975. These panels included such relays as GE CFVB, GE CEH and GE IJD.

Thus 7 relay types are qualified for FPL's low seismic demand. This leaves 4 relay types to evaluate after Step 2 is completed.

Step 3. For the remaining relays, other qualification information such as previous responses to I & E notices may be sufficient to demonstrate the relay's capability or the fact that it is not used in Safety Related circuits. As an example, I & E Notice 88-14 identified as a concern the use of normally de-energized, normally closed contacts of HGA relays. The notice was reviewed and our review efforts determined that only HGA II and HG III relays were a concern in the notice, and these relays are not used in this manner in safety related equipment at FPL.

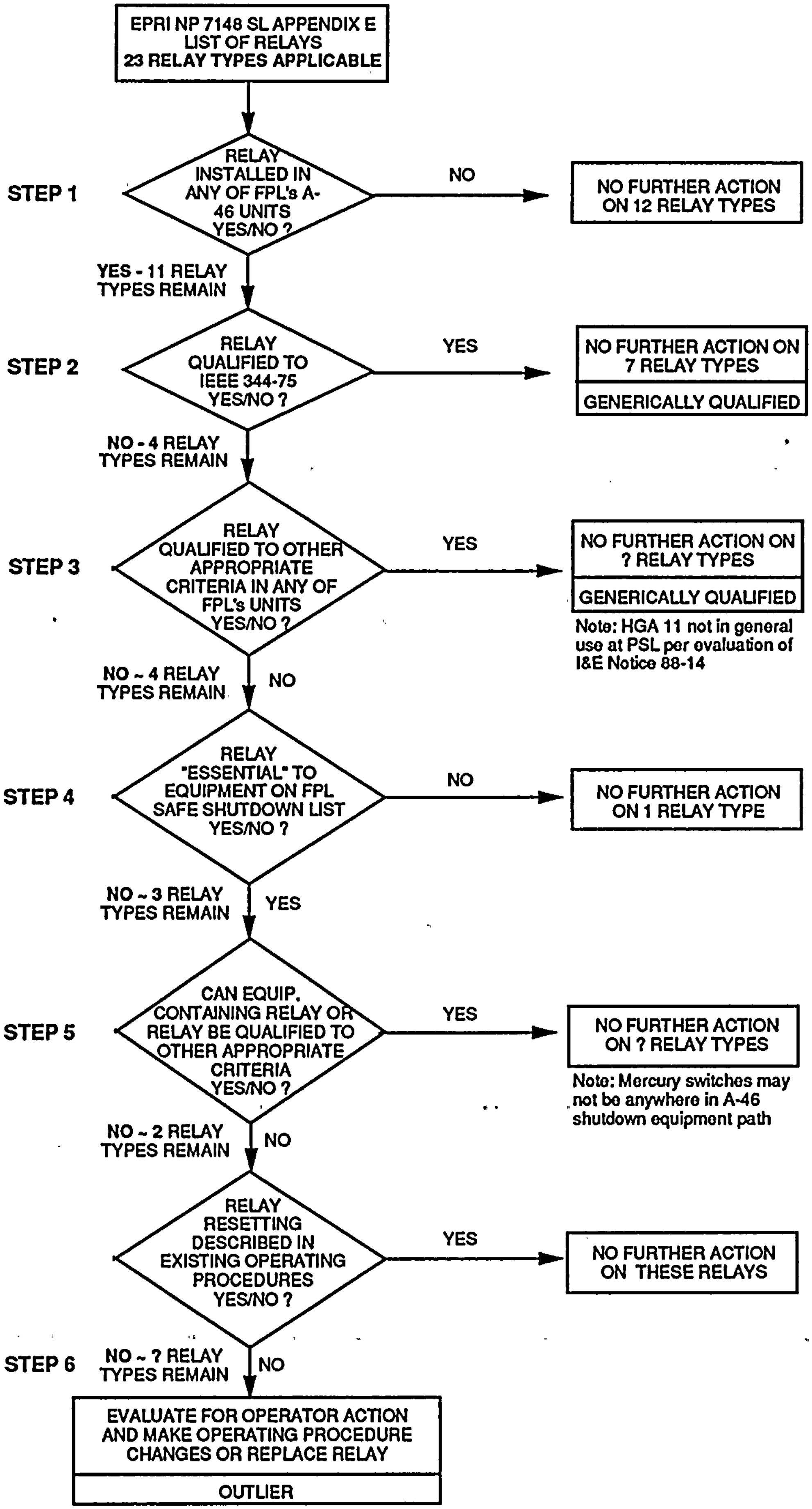
Step 4. Relays not essential to equipment on FPL safe shutdown list. Westinghouse HU relays are high speed single phase relays used in the differential protection of transformers. They are applied where magnetizing inrush current to a transformer is severe. Typically, they are used to protect large transformers such as the Generator Main Transformers. Hence, we do not expect to find any in the A-46 equipment shutdown path.

Step 5. Mercury switches are not generally used in safety applications due to their mercury content and potential contact chatter due to vibrations. Hence, we do not expect to find any in the A-46 equipment shutdown path.

Step 6. Any relays not qualified in the previous steps are considered to be "outliers" and appropriate procedural action would be taken or the relays would be replaced. If existing operating procedures or procedure changes can safely address potential chattering of "outlier" relays, they will not need to be replaced. However, if a major engineering design change to the equipment is made in the future, IEEE 344 qualified relays would be used as replacements.

The foregoing provides examples from our preliminary investigation of the A-46 relay issue and while it does not provide the final results of our investigation, it serves as a sample and explanation of the process. Once this methodology is found acceptable by the NRC, a complete and thorough review will begin.

FPL A-46 RELAY EVLAUATION PROGRAM



ATTACHMENT C

RESPONSE TO THE NRC REQUEST FOR ADDITIONAL INFORMATION (RAI) DATED JUNE 23, 1993

1. **Question:**

Item No. 5 of the March 18, 1992 RAI requested a listing of the safe-shutdown path equipment and equipment categories, (including those inside containment, which are part of the safe-shutdown path, but which have been excluded from the seismic adequacy verification program) and detailed technical justification for the exclusions unless they were in accordance with GL 87-02. In response, you provided Piping and Instrument Diagrams, along with associated equipment lists, which identify both the items included in the USI A-46 walkdowns and all items which were excluded (Enclosure A to your September 8, 1992 letter). You stated that the exclusions consist of equipment which are either not within the lesser scope of FPL's program, passive items, or items excluded by the conditions set by GL 87-02.

Although you have identified passive equipment items, you have not provided clear justification for excluded equipment (e.g., there were no valves included in the walkdowns). Excluding an equipment item solely on the basis that the item is not within the lesser scope of FPL's program is not acceptable. (Note: We recognize that the Auxiliary Feedwater System was previously inspected as part of GL 81-14, and was, therefore, exempt from further review).

Please provide specific technical justification for excluding individual pieces of equipment.

Response:

Equipment included in the lesser scope of FPL's program was chosen on the basis of its contribution to seismic risk and is explained further as follows.

On June 2, 1988, FPL made a presentation to the NRC at its White Flint offices on the Subject of GL 87-02. The presentation included Value/Impact analyses, using the NRC's methodology in NUREG 1211, which showed that the Value to FPL of resolving all of the potential concerns in GL 87-02 was approximately $10,000 per nuclear unit. The presentation also showed the low seismicity and low seismic hazard associated with FPL's nuclear sites. Subsequent to the White Flint meeting, the NRC wrote FPL on June 16, 1988 stating that a resolution program for GL 87-02 "scaled back" from the program described in the generic letter would be acceptable. The NRC further stated that if FPL did not want to submit a "scaled back" program, it could file an appeal with NRC management.

In FPL's opinion, the larger contributors to seismic risk included anchorages of tanks and equipment and these were included, even though the costs exceeded the value of $10,000 previously referred to. As lesser risk contributors were reviewed together with the costs associated with their inclusion in the program, it became evident that their inclusion could not be justified on the basis of Value/Impact and that it would not be prudent to do so. FPL's program was therefore scaled back from the program suggested by GL-87-02 (Rev. 0) using a combination of contribution to seismic risk and Value/Impact prudency considerations.

2. **Question:**

Your equipment capacity vs. demand evaluations were performed on a plant-specific basis using the Senior Seismic Review and Advisory Panel (SSRAP) Bounding Spectrum instead of an equipment-specific basis. In order to use the SSRAP Bounding Spectrum for representing an equipment item's seismic capacity, you <u>must</u> demonstrate, on an equipment-specific basis, that each of the equipment items, for which a seismic capacity vs. demand evaluation is required, is similar to an equipment category within the scope of the experience data base. Provide criteria and procedures, or references for this demonstration.

The SSRAP Report, Revision 4 dated February 28, 1991, provided partial descriptions of the experience data base equipment categories and includes several caveats and exclusions for each equipment category. Our review of the Seismic Qualification Utility Group's Generic Implementation Procedure (GIP-2) for resolving USI A-46 has concluded that the equipment descriptions (and the caveats and exclusions) provided in the SSRAP report were not complete. For acceptable descriptions of equipment categories for using the experience data base, refer to Appendix B of GIP-2 as corrected on February 14, 1992, and as supplemented by the staff's Supplementary Safety Evaluation Report No. 2 dated May 22, 1992.

Response:

Seismic Capacity vs. Demand was evaluated at the plant level. It was judged that since the design basis earthquake levels at FPL's sites were significantly lower than the SSRAP Bounding Spectra, that equipment was located at relatively low elevations in the plants, and that equipment was located in typical nuclear structures, the amplified response of all components at the FPL plants to a design basis event is significantly less than what the equipment in the data base plants experienced during real strong motion earthquakes. Therefore, it was not necessary to compare Capacity vs. Demand for each equipment item.

The equipment included in the walkdown at the FPL plants is similar to the equipment at the other A-46 plants throughout the industry and the earthquake experience data base. This was confirmed during the walkdown by the expert SRT, and a subsequent review (size and weight) of all walkdown data sheets by Stevenson & Associates.

The best reference available to use as walkdown guidance at the time of FPL's walkdowns was the SSRAP Report, Revision 4, dated February 28, 1991. At that time, all major issues with regard to seismic adequacy of equipment and equipment performance during earthquakes were well defined. These major issues were included in the Rev. 4 SSRAP report. The SRT was well aware of these issues having either authored, developed or reviewed all of the documents available on the subject of earthquake experience and seismic walkdowns. FPL believes that the SRT, because of their in-depth knowledge of earthquake experience, has superior skills at seeking out any detail that was seismically vulnerable at the FPL plants. It is the FPL position that the equipment and caveat descriptions in the draft Rev. 4 SSRAP report augmented by the SRT knowledge is acceptable for addressing equipment construction adequacy (referred to in the FPL reports) or equipment categories and specific caveats (referred to in this RAI question).

3. Question:

In response to the March 18, 1992 RAI, Item 9, you indicated that evaluation of the seismic adequacy of the above-ground vertical tanks is included in your USI A-46 implementation program in order to resolve USI A-40. However, the adequacy of ring foundations for the steel tanks is not addressed in your USI A-46 program. The provision for checking the adequacy of tank-foundations is a part of the resolution of USI A-40, as reflected in Section II.14.i of SRP 3.7.3 dated August, 1989.

Please provide information relating to the adequacy of all of the applicable tank-foundations.

Response:

Above-ground vertical tanks at the Turkey Point Units 3 and 4 plants are founded on mat foundations. Three tanks considered at St. Lucie Unit 1 (the Diesel Oil Storage Tank, Refueling Water Storage Tank, and Condensate Storage Tank) have ring foundations. At the time of the walkdown, no difference in consideration between ring and mat foundations had been identified since the SRT were not aware of any earthquake experience data which indicated any difference in the seismic response of these two types of tank foundations. Review by the SRT for these tanks was limited to the capacity of the anchor bolts and anchor bolt chairs. At the request of the

SRT, conservative calculations to check these items were performed with the overturning moment calculated neglecting the fluid holddown force. The minimum factor of safety obtained was about 6. For ring foundations, GIP Rev. 2, Section 3.3.7 indicates a check of the rebar in the foundation for adequacy to resist the overturning should be made. An additional calculation performed for this purpose indicates a minimum factor of safety of 9.6 for the three St. Lucie tanks identified if fluid holddown forces are neglected. It should be noted that inclusion of the fluid holddown forces indicates no overturning would occur, and hence no tension in the anchor bolts or uplift on the foundation would result.

4. Question:

In response to Item No. 11 of the March 18, 1992 RAI, you indicated that with the exception of the torque tightness, your equipment anchorage criteria would be in accordance with Volume 1 of EPRI NP-5228, Revision 1, dated June, 1991. Torque tightness checking is one of the most important attributes in ensuring the integrity of the bolted expansion anchorages. You did not provide any justification for excluding torque tightness testing from your program. Further, your Final Report references a preliminary version of EPRI-NP-5338 dated May, 1987, and it is also referenced in many of the Appendix C supporting calculations. We have not endorsed the preliminary version of the EPRI report.

Please describe your methodology to ensure torque tightness checking of equipment anchorages. Note that a random sample testing of the affected anchorages would be acceptable.

Please revise your Final Report to reflect compliance to the final version of EPRI NP-5228, Revision 1.

Response:

The walkdown and supporting calculations were performed prior to the publication of Volume 1 of the EPRI NP-5228, Revision 1. Therefore, this publication is not referenced in the FPL report. However, the SRT was aware of substantive changes to be published in the Rev. 1 document due to their unique knowledge of these publications as described in Response to RAI Question 2. The walkdown procedure developed by FPL utilized a non-intrusive walkdown by a team of experts. Although bolt tightness checking may be considered as "one of the most important attributes to ensuring the integrity of the bolted expansion anchorages", FPL is not aware of any data that exist comparing the capacity of bolts failing this test to bolts passing the test. FPL did not include the bolt tightness check in the procedure, because of its intrusive nature.

Although not available in final form at the time of the walkdowns, a systematic review was conducted by Stevenson & Associates of the FPL walkdown data sheets and calculations for compliance with the final version of report EPRI NP-5228, Revision 1. There were no changes to the conclusions reached during the walkdown as a result of this review. The final report will be changed to reference this review and also include Revision 1 of EPRI NP-5228 as a reference.

5. **Question:**

Final Report Section 4.2.3 indicates a factor of safety (FS) of 3.0 for all expansion anchors. This position is inconsistent with EPRI NP 5228, Revision 1, which in some cases specifies an FS greater than 3.0 for expansion anchors, such as in cracked concrete.

Please justify the use of a single FS of 3.0 for all expansion anchors.

Response:

The factor of safety of 3 was used for expansion anchorages that were not subject to reductions (i.e. cracked concrete, etc.) as described in EPRI NP-5228 Revision 1 (as confirmed by a systematic review of the walkdown data sheets and calculations).

6. **Question:**

With respect to the evaluation of equipment anchorages during the walkdowns, Final Report Section 4.2.3, page 19, states "When the anchorage was obviously rugged (Seismic Review Team) SRT judgment was used to asses anchorage adequacy." This judgment was performed in the context of the above criteria.

Please identify the specific criteria used to make such judgments and clarify whether the anchorage criteria is per EPRI NP-5228, Revision 1.

Response:

As discussed in response to RAI question 4, the walkdowns were performed prior to the publication of Volume 1 of the EPRI NP-5228, Revision 1. However, the SRT was aware of substantive changes to be published in the Rev. 1 document due to their unique knowledge of these publications as described in Response to RAI question 2. The screening judgments of obviously rugged equipment anchorages were made in the context of the existing publication and this knowledge.

Also, although not available in its final form at the time of the walkdowns, a systematic review was conducted by Stevenson & Associates of the walkdown data sheets for the screened

equipment items for compliance with the final version of report EPRI NP-5228, Rev. 1. There were no changes to the conclusions reached during the walkdown as a result of this review.

7. Question:

Final Report, Appendix A, Walkdown Procedure, page 6, discussed two levels of "screening out" equipment anchorages during walkdowns. For the first screening, you state that the anchorage was evaluated to determine if it is in conformance with the design basis for the plant. Your seismic design basis discussion (Final Report, Section 3.4, page 9) indicated that there were no specific comments with regard to the seismic design of anchorages for mechanical and electrical equipment.

Please explain in detail how the first "screening out" of equipment anchorage was performed (e.g. was the anchorage screening out based on the criteria in NP-5228, Revision 1?).

Response:

In order for equipment to be screened for anchorage, it met both screens described in the FPL Walkdown Procedure. The first "design basis" screen referred to in the procedure is the existing plant anchorage drawings when they were available. FPL restored equipment to the "as designed" configuration regardless of whether or not the anchorage could meet the EPRI NP-5228 criteria for the seismic demand for the plants. The second anchorage screen referred to in the procedure were those anchorages that were judged "obviously rugged" by the SRT. The specific criteria used to meet these judgments are described in the response to RAI question 6 above. The next level of screening that was used for equipment items whose anchorage was not "obviously rugged" were calculations using the same criteria.

8. Question:

Please justify the assumption of concrete compressive strength greater than or equal to 3500 psi used in the anchorage calculations. Note that for concrete strengths below 3500 psi, EPRI NO-5228, Revision 1 requires application of a capacity reduction factor.

Response:

The minimum concrete compressive strength at the FPL plants is 3000 psi. Review of the supplemental calculations performed indicates that for three generic anchorage calculations performed for St. Lucie Unit 1, a value of 3500 psi was assumed. Review of these calculations using the value of 3000 psi indicates no effect on one tank, a reduction of 3% and 7%

of the indicated tensile capacities on another tank, and a reduction of 18% for pullout capacity on the third tank. However, these reductions do not alter the acceptable conclusions. All other calculations for Turkey Point Units 3 and 4 and St. Lucie Unit 1, for which concrete strength was considered, used 3000 psi or less.

9. **Question:**

The equipment data sheets (EDS) completed by the SRT do not specifically address many of the anchorage concerns discussed in EPRI NP-5228, Revision 1. The EDS do not specify checks for such items as anchor spacing, free-edge distance, gaps under bolted anchorage, effect of prying action, etc.

Please explain how the capacity reduction factors for closely-spaced anchors, near-edge anchors, cracked concrete, etc., which are described in EPRI NP-5228, Revision 1, were considered during the walkdowns and in the anchorage evaluations.

Response:

When the capacity reduction factors were applicable for closely-spaced anchors, near-edge anchors, cracked concrete, etc., they were considered. The systematic review of the walkdown data sheets and calculations for compliance with the final version of report EPRI NP-5228, Revision 1 confirmed this.

10. **Question:**

Final Report, Reference 13, requires comparison of applicable Instructure Response Spectra (IRS) with 1.5 times the bounding spectrum.

Please provide documentation of these comparisons for the critical equipment.

Response:

For St. Lucie Unit 1, the only component in our equipment list above the 40' elevation was the Component Cooling Water Surge Tank, which is located approximately 53 feet above grade. The seismically vulnerable components of this item were evaluated using the applicable floor response spectra instead of the 1.5 bounding spectra comparison as further discussed in the St. Lucie final report section 4.2.1.

For Turkey Point, the only component in our equipment list above the 40' elevation was the Component Cooling Water Surge Tank, which is located approximately 52 feet above grade. The seismically vulnerable components of this item were evaluated using a conservative estimate of the floor response as further

discussed in the Turkey Point final report section 4.2.1.

11. **Question:**

On the basis of the IRS, (Section 4.2.3 Final Report) for assessment of anchorages, you propose to use 0.5G and 1.2g demands for equipment natural frequencies above 4.5 Hz, and below 4.5 Hz, respectively. However, for equipment where IRS is not available, you propose to use a horizontal acceleration of 0.3g per SSRAP.

Please provide justification for use of different demand values in addressing the anchorage capacity.

Response:

For equipment items located about 40 feet above grade, and having a natural frequency above 8 Hz, seismic demand to assess anchorage (as recommended by SSRAP, and approved by the USNRC in the Safety Evaluation Report on the Rev. 2 GIP dated May 22, 1992), can be taken as the peak of the 5% damped site ground response spectrum times a factor of 1.5 to adjust to a median centered value times an additional 1.25 modification factor. For the FPL plants, this results in a value of .3g arrived at by taking .16g (5% ground response peak) x 1.5 x 1.25. The demand can also be taken from In-structure Response Spectra (IRS) if available. This was the method used on the FPL plants. Due to the fact that 5% equipment damped IRS were not available from the existing IRS for the FPL plants, a lower damping (2%) was conservatively used which results in seismic demand values of .5g being established for equipment with fundamental natural frequencies > 4.5Hz and 1.2g for equipment with fundamental natural frequencies < 4.5Hz.

12. **Question:**

The adequacy of the above-ground vertical tanks has been assessed by considering all anchor bolts to be subjected to varying amounts of tension loads and the small area of the tank shell resisting the compression load.

Please provide an assessment of the buckling mode of failure of tank shells. Also provide an assessment of shear stresses on anchor bolts and tank shells considering the vertical component of the postulated earthquake..

Response:

The calculations for the St. Lucie tanks included a methodology that restricted the outermost bolt (the bolt with the most tension) to 34 ksi (below yield). The remaining bolts were assumed to have a tension load of zero at the compression edge of the tank and varying linearly between 0 and 34 ksi between. In order for the tank shell to buckle

under compression loads, significant uplift of the tank and bolt yielding must occur. By meeting the criteria described above and having an additional factor of safety of capacity compared to demand, buckling cannot occur.

The calculations for the tanks at Turkey Point used Dr. Kennedy's High Confidence Low Probability Failure (HCLPF) methodology contained in NUREG/CR-5270. This methodology includes the buckling mode of failure as the limiting failure mode.

The assessment of shear loadings on the tank considering the vertical component of the postulated earthquake are included in the calculations. In all cases, the shear was resisted by friction at the base of the tank without the need of including additional stress on the anchor bolts. This approach for assessing shear is consistent with the approach approved by the USNRC in the Safety Evaluation Report on the Rev. 2 GIP dated May 22, 1992.

13. Question:

Please demonstrate the adequacy of the safety-related electrical raceways (cable trays and conduits) and their supports, which is within the scope of the original GL 87-02. Provide your plans for resolving this item of the Generic Letter.

Response:

Seismic experience with raceways has demonstrated that they perform very well in seismic events, even raceways with no seismic design in normal commercial/industrial construction, and consequently they are small to negligible contributors to seismic risk. Raceways were among the lesser risk contributors which were reviewed for possible inclusion in FPL's "scaled back" program. They were omitted because there was no apparent cost benefit to the safety of the plants.

14. Question:

In response to Item 8 of the March 18, 1992 RAI relating to the development of IRS, you believe that providing this information is not within the scope of USI A-46. However, we understand that you would be willing to discuss this information as divorced from the resolution of USI A-46. To assure that proper procedures are used in verifying the adequacy of the equipment, we plan to discuss this issue during future audits of the St. Lucie Unit 1 and Turkey Point Units 3 and 4 USI A-46 programs.

Response:

The NRC and SQUG agreed to include the devlelopment of the IRS in the resolution of A-46 for SQUG utilities although it was never in GL 87-02 (Rev. 0). FPL has never committed to include the development of the IRS in the resolution of A-46; however, FPL would be willing to discuss this information with the NRC, but not on the A-46 docket.

15. Question:

Please revise the Final Report to delete references to ACI 349, Appendix B, relating to anchorage criteria as per your response to Item No. 12 of the March 18, 1992 RAI.

Response:

The final report will be changed to delete this reference.