January 31, 2001

Mr. T. F. Plunkett
President - Nuclear Division
Florida Power and Light Company
P.O. Box 14000
Juno Beach, FL 33408-0420

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RELATED TO THE STAFF'S REVIEW OF SEVERE ACCIDENT MITIGATION ALTERNATIVES FOR TURKEY POINT UNITS 3 AND 4 (TAC NOS. MA9940 AND MA9944)

Dear Mr. Plunkett:

The NRC staff has reviewed Florida Power and Light Company's analysis of severe accident mitigation alternatives (SAMAs), submitted as part of the application for license renewal for Turkey Point Units 3 and 4. The staff has identified areas where additional information is needed to complete its review. Enclosed are the staff's requests for additional information (RAIs).

As discussed with your staff, we request that you provide your responses to these RAIs within 60 days of the date of this letter in order to support an accelerated review schedule. If you have any questions, please contact me at (301) 415-1108.

Sincerely,

/RA/Signed by:JHWilson

James H. Wilson, Senior Project Manager
Generic Issues, Environmental, Financial, and
Rulemaking Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation.

Enclosure: As stated

cc w/encl: See next page

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OFFICE OF NUCLEAR REACTOR REGULATION
REQUEST FOR ADDITIONAL INFORMATION
RELATED TO THE STAFF'S REVIEW OF
SEVERE ACCIDENT MITIGATION ALTERNATIVES
RELATED TO LICENSE RENEWAL FOR TURKEY POINT
UNITS 3 AND 4 (TAC NOS. MA9940 AND MA9944)

1. In the Severe Accident Management Alternative (SAMA) analysis reported in Reference [1], the base case Core Damage Frequency (CDF) appears to be based on the internal events CDF that is estimated from the current PRA model (which is a modification to the original Individual Plant Examination (IPE) [2] that was reviewed by the U. S. Nuclear Regulatory Commission (NRC) [3-5]). Reference [1] reports several different numbers as the base case CDF, namely:

   On Page 4.20-11, the reported CDF is 1.62×10^{-5} per reactor year.

   On Page 4.20-18, the reported CDF is 6.12×10^{-5} per reactor year.

   The sum total of the release frequencies over all possible release modes (i.e., if all the release modes are considered, this should be the same as the total CDF) listed in Table F.1-2 (on Page F.1-4) is about 9.14×10^{-6} per reactor year.

   On the other hand, the internal events CDF from the original IPE was 3.7×10^{-4} per reactor year [2-3], whereas the CDF following the plant modification resulting from the IPE process was 1.0×10^{-4} per reactor year.

   Please provide the following:

   a. The correct internal events CDF that is the basis for the present SAMA evaluation [1], including the detailed rationale behind the use of the CDF estimate in this type of analysis for license renewal. Specifically, please provide the reasons for the reduction in the internal events CDF from the IPE estimate of 1.0×10^{-4} per reactor year, to the current level (1.62×10^{-5}, 6.12×10^{-5}, etc.).

   b. In the original IPE, the Steam Generator Tube Rupture (SGTR), and the Interfacing System LOCA (ISLOCA) were found to contribute each, about 4 percent (or about 4×10^{-6} per reactor) [2-3], to the total internal events CDF. In Table F.1-2, the core damage frequencies for SGTR (i.e., BP-SGTR), and ISLOCAs (i.e., BP-V) are listed as 1.71×10^{-8} and 6.24×10^{-8} per reactor year, respectively. Please provide an explanation for the reductions of about a factor

of 60 for ISLOCA and 250 for SGTR (i.e., were these reductions due to changes in procedures or inspection programs, or as a result of modifications to the PRA parameters?).

c. Please provide justifications for the seemingly low releases shown in Table F.1-2 for SGTR release mode. This should include either detailed plant-specific results of calculations and/or generic results for other plants, including a documentation of their technical bases. Please show the sensitivity of conditional consequences resulting from SGTR scenarios if accident source terms more representative of other published PRAs (e.g., NUREG-1150) were to be used in the SAMA analyses.

d. Depending on the correct value of total CDF, the contribution to late containment failure for scenarios for which the containment sprays are operational (i.e., release modes C3-L and C3-R) ranges from 3 percent to 23 percent. Please list the main reasons for late containment failure when the containment sprays are operating. This issue was identified as an inconsistency in the NRC review of the original IPE submittal (see Page 20 of Reference [4]).

e. Due to inconsistencies in the reported base-case CDF, it is difficult to determine the number of accident sequences that have been included in the level-3 consequence (i.e., Table F.1-4) and risk calculations. Please confirm the scope of accident consequence calculations in terms of the core damage sequences and the radiological release modes.

2. According to the original IPE [2], the transient-induced loss of coolant accident (LOCA) resulting from the reactor coolant pump (RCP) seal failure contributed almost 60 percent to the total internal events CDF. As a countermeasure against the risk-dominant sequence of RCP seal LOCAs, plant modifications were introduced. The SAMA analysis documented in Reference [1] is believed to be based on the risk profile following the IPE-based plant modifications. To support using the updated risk model in the SAMA identification and evaluation processes, please provide the following.

   a. A description of the level-1 and level-2 risk profiles, results, and insights in terms of the <u>major contributions</u> (hardware and human failures) to the core damage frequency and release frequencies following the IPE-based plant modifications.

   b. A specific discussion of the major differences between the SAMA PRA as compared with the original IPE, explaining any plant and/or modeling changes that have resulted in the new CDF and release frequencies. What quality programs are in place to ensure that the plant modeled in the SAMA PRA is consistent with the as-built configuration of the plant? In addition, please provide a description of any internal and external peer review of the latest level -1, -2, and -3 portions of the PRA.

c. A list of key equipment failures and human actions that dominate CDF and the large early and late release frequencies, which have the greatest potential for reducing the risk of severe accidents at Turkey Point, along with the results of any supporting importance analyses (e.g., Fussel-Vesely and/or risk reduction importance measures).

3. Risk analyses at other commercial nuclear power plants indicate that external events could be large contributors to core damage and the overall risk to the public. The Turkey Point IPE [2] has estimated the CDF for control room fires to be as high as 1.9×10^{-4} per reactor year, and the CDF for the storm surges[1] at the site ranging from 1×10^{-4} to about 1×10^{-6} per year. Even though Reference [2] claims that there are conservatisms associated with the control room fire-induced CDF, the fire and other external events (e.g., storm surge events) have contributions that are of the same order of magnitude as the internal events CDF. The quantitative influence of SAMAs applicable to internal fires/floods and external events have been evaluated by just doubling the estimated internal events CDF. Specifically:

a. In view of the fact that the characteristics of the internal and external events scenarios are, in general, considerably different, please demonstrate (i.e., through sound PRA arguments), considering the uncertainties in PRA results, that by doubling the internal events CDF, one can reliably bound the risk of core damage due to all initiators at Turkey Point.

b. Provide a justification for including only a very limited number of SAMA candidates that involve external and other events (i.e., one seismic event [SAMA 150], one tornado event [SAMA 164], two candidates address internal flooding events [SAMAs 99 and 100]). In this regard, please discuss how plant-specific external and other CDF initiators (e.g., fires, flood, storm surges, etc.) were considered in the SAMA identification and assessment process. Furthermore, how was the quantitative impact of SAMA 99 explicitly assessed?

4. As described in Generic Letter 88-20, an important objective of the IPE/IPEEE program was to identify plant-specific vulnerabilities to severe accidents. The IPE study for Turkey Point Units 3 and 4 identified the following items of potential enhancement to the plant's accident management capability:

- Replenishment of Refueling Water Storage Tank (RWST)
- Primary System Depressurization
- AC Power Recovery
- Cross-connection of Component Cooling Water (CCW)
- Manual Actuation of Containment Spray (Cavity Flooding)

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[1] The Turkey Point IPE [2] concluded that the storm surge dominates all other hurricane hazard component.

In this regard, please discuss the potential design enhancements and procedural modifications identified through the Turkey Point IPE, IPEEE and any other follow-on studies, and the disposition/status of these items. For those that have not been implemented[2], please provide your results of their assessment within the context of SAMAs, showing their potential viability for implementation within your risk management program. Also please discuss how the insights gained from examination of these potential improvements were addressed in the SAMA identification process.

In the staff SE (Reference [3], Pages 12-14), it is pointed out that your original IPE conditional failure probability for late containment failure was 62 percent. Subsequent analysis, in response to staff questions, took credit for certain recovery actions and conservatisms. Your new value for the conditional failure probability for late containment failure was 7 percent. Please describe the procedures and/or hardware fixes that allow for the recovery actions. Do you have an estimate of the reduction in risk that resulted from such recovery actions? Are there opportunities for SAMAs to further reduce the containment failure probability? (Although the staff recognizes that many “containment” candidate SAMAs are excessively expensive [such as SAMA 46], there are others that are much less expensive and can effectively mitigate the consequences of some core damage accidents [for example passive autocatalytic recombiners for hydrogen control cost about $40,000 per recombiner].)

Some of the reduction from 62 percent to 7 percent was attributed to the removal of conservatisms. (The staff noted that some of the assumptions “appear somewhat optimistic.”) In light of this, what uncertainties do you associate with the risk and risk reduction values that make up your SAMA assessment?

In the ER (Page 4.20-20), you state that “…an expert panel reviews the benefit to determine whether the SAMA can be implemented for a cost equivalent to twice the benefit.” Please provide the cost determination for a sampling of procedural and hardware SAMA candidates, which were close to your “twice the benefit” guideline. For example, you screened out candidate SAMA 47, “Use fire water spray pump for containment spray.” What are the cost and benefit estimates for this SAMA?

5. In the SAMA study [1], the basis for the final SAMA screening and cost-benefit analysis is not clearly spelled out. For example, the screening analysis of SAMA 155, “Provide a centrifugal charging pump”, is based on the risk impact analysis such that the charging pump failures have less than a 2.5 percent contribution to the internal events CDF. Please provide detailed description of the risk impact analysis, including the specific value of ΔCDF and Δperson-rem, for each of the final SAMA candidates.

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[2] For instance, flooding of a failed steam generator following a SGTR event has been considered as a potential severe accident mitigation measure in severe accident mitigation studies for other plants.

6. The base-case CDF used in the SAMA evaluation, and the contribution of various release modes have changed significantly since the original IPE [2] was completed. For instance, based on the information presented in Reference [1] (e.g., Table F.1-2) one cannot determine if the changes in the risk contributions only result from changes in the estimated CDF (i.e., plant damage state frequencies), or changes in containment performance analyses. Therefore, to better assess the technical basis for the base-case, please provide the following:

   a. A list of all plant damage states, their definition <u>where different</u> from original IPE [2], their respective frequencies based on the latest PRA results used in the SAMA submittal.

   b. A containment matrix based on the latest PRA (i.e., similar to Table 4.6-30 of Reference [2]), <u>if different</u> from the original IPE as reported in Reference [2]. In addition, please justify any changes in the estimated conditional probabilities associated with each release mode, given the plant damage states (i.e., the various end-state probabilities in the containment event trees) since the original IPE [2] was completed and reviewed by NRC [3, 4]. These changes should be supported by specific information on the advances in the state-of-the art, computer code calculations, experiments and any other relevant data.

   c. A list of all changes to the level-2 PRA assumptions and/or models that could impact the level-2 results (e.g., item b above) should be provided.

7. Please provide the uncertainty range associated with the internal events CDF for the base-case PRA. Please show the impact of considering the uncertainties in the estimated CDF on the current SAMA cost-benefit conclusions. If the uncertainties in estimated CDF are not available, please determine the qualitative risk impact by considering the impact of uncertainty ranges as reported in typical level-1 PRA (e.g., NUREG-1150), on the estimated risk results.

8. On Page F.1-6 of Reference [1], it is stated that, “For all modes the RU, LA, CE, and BA fractions of the usual MACCS2 species are set to zero, as they were not reported in the Individual Plant Examination (IPE) submittal.” Please perform a sensitivity calculation for several risk dominant release modes (e.g., based on the frequencies listed in Table F.1-2, and the consequences in Tables F.1-4 and F.1-5, the late containment failure release classes C4-L and C4-R appear to be among the risk dominant release modes for the base case analyses; however, other release modes that also have a major contribution to the total base case risk should be included also), by allowing for appropriate releases for these refractory species consistent with similar release categories in Surry NUREG-1150 or other published studies (e.g., Table 4.7-2 of Reference [2] provides examples of several applicable source term studies that contain releases for more complete set of radionuclide groups), to demonstrate that the risk impacts justify this assumption in Reference [1].

9. Please list the contribution of each release mode (Table F.1-2), in terms of the percentage of the total annual risk of population dose and offsite economic cost in a tabular format similar to Tables F.1-2 and F.1-4). Also, please list the total annual risk of population dose (i.e., Sieverts per year) and offsite economic cost (i.e., dollars per year). In addition, for each release mode, please provide a table listing the release energy, release duration, the evacuation warning time, assumed shielding factors for buildings/shelters and justifications for these selections.

10. Table F.1-2 lists the core inventory for the uprated power of 2300 MW(t) for Turkey Point based on the scaling of the MACCS end-of-cycle inventory for a 3412 MW(t) plant. Please justify (i.e., by providing plant-specific data or ORIGEN or other equivalent plant-specific computer code calculated burn-up results) that the power-scaled standard MACCS core inventory as used in Reference [1] is applicable to the present-day Turkey Point fuel burn-up histories for typical Turkey Point end-of-cycle conditions.

References

1. "Applicant's Environmental Report Operating License Renewal Stage" Turkey Point Units 3 & 4, Florida Power and Light, Docket Nos. 50-250 and 50-251, Revision 1.

2. "Turkey Point Plant Units 3 & 4 Probabilistic Risk Assessment - Individual Plant Examination," Final Report, June 25, 1991.

3. "Staff Evaluation of Turkey Point Individual Plant Examination (IPE) (Internal Events Only)," Enclosure 1 to letter from L. Raghavan (NRC) to J. H. Goldberg, Florida Power and Light, October 15, 1992.

4. J. Meyer, M. Khatib-Rahbar, and R. Vijaykumar, "Turkey Point IPE Back-End Audit," Task 3 Technical Evaluation Report, SCIENTECH, Inc. SCIE-NRC-207-91, Rev 1, December 21, 1991.

5. R. T. Sewell, et al., "Technical Evaluation Report on the "Submittal-Only" Review of the Individual Plant Examination of External Events at Turkey Point Nuclear Plant, Units 3 and 4," Energy Research, Inc. ERI/NRC 95-507, January 1998.

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