October 24, 1991

Docket Nos. 50-250
and 50-251

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Mr. J. H. Goldberg
President - Nuclear Division
Florida Power and Light Company
P.O. Box 14000
Juno Beach, Florida 33408-0420

Dear Mr. Goldberg:

SUBJECT: TURKEY POINT UNITS 3 AND 4 - INDIVIDUAL PLANT EXAMINATION (IPE) STAFF REVIEW

As part of our continuing review of the Turkey Point IPE submittal and its associated documentation, the review team intends to visit the Turkey Point site from November 19 through November 21, 1991. The enclosure provides our initial list of questions related to the internal event analysis and response to the containment performance improvement (CPI) program recommendations. These questions will be discussed with your staff during the site visit and many of these questions can be resolved at that time. However, following the site visit, additional information may be necessary in order to complete our safety evaluation. The visit is being coordinated with your licensing staff.

If you have any questions, please call me at (301) 492-1471.

Sincerely,

|s|
Rajender Auluck, Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
Questions

cc w/enclosure:
See next page

LA:PD22	PM:PD22	D:PD22
DMiller	RAuluck	HBerkow
10/23/91	10/23/91	10/23/91

Document Name: LETTER TO TURKEY POINT

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ENCLOSURE

QUESTIONS ON TURKEY POINT IPE SUBMITTAL

FRONT-END

A. Issues involved in accident sequence modeling

1. Please discuss the process used to link initiator (e. g., loss of instrument air, loss of service water, loss of a DC bus, loss of a 4.16 KV bus, loss of HVAC), and partial loss of function, (e. g., room cooling for charging pumps, RHR pumps, and TDAFW pumps) with associated fault tree (FT) and applicable sequence tree or equivalent. The discussion should include the impact of DC load shedding, isolation of non-essential heat loads, and backup air bottles, as applicable.

2. Discuss Turkey Point Plant's (TPP's) treatment of mutually exclusive events and their impact on downside risk (core damage frequency).

3. Discuss the process used to apply component (e.g., MOVs) beta factors during the sequence quantification process.

4. Provide a brief discussion related to the quantification of interfacing LOCA scenarios (Ref. NUREG/CR-5606) including applicable data used for MOVs.

5. Discuss treatment of LOCA sequences (of all sizes) with the containment isolation failures including the impact on peak clad temperature.

6. Discuss the quantification of the steam generator overfilling scenarios (Ref. NUREG/CR-4385) and resulting decay heat removal sequences.

7. Discuss the quantification of the main control room (MCR) critical instruments and alarms during a time phased station blackout scenario.

8. Discuss the impact of early initiation of the containment spray system on the decay heat removal system reliability.

9. For significant scenarios, discuss the concept of "Near Miss Core Damage" concept for the TPP facility. Such discussion should focus on plant-specific information on the very last single action (by one individual or a group of individuals) to be performed on a piece of hardware to avoid a core damage event following an initiating event. It should be noted that such information could be obtained through a careful review of the associated information regarding component failure modes and operator errors (both omission and commission) contained in the cutsets of all sequences (dominant and non-dominant).

B. Issues involved in sensitivity and/or uncertainty analysis

1. Reference has been made to the NUREG-1150 risk analysis in assigning the uncertainty to each parameter analyzed in the TPP PSA. However, the NUREG-1150 method has not been used in selecting the appropriate sensitivity parameters (e. g., ECC pump room cooling based on a detailed calculation). Thus, provide a discussion of the process used to select the appropriate parameters or combination of parameters (e. g. the impact of high MOV failure rates, the impact of high MOV and check valve failure rates) for a detailed sensitivity analysis, and the method of updating the overall TPP database for a final sequence quantification process.

2. Discuss the process used to identify failure modes that could lead to total pressure drop across the RCP seals and provide the results of a sensitivity analysis of the above item on the overall core damage frequency estimates.

C. Issues involved in plant-specific data analysis

1. The plant-specific data collected for the TPP facility and documented in Section 3.3 and 3.4 are noteworthy. Please discuss the process used to estimate the beta factors (documented in Table 3.3-6). Also discuss the beta factor (or other method) associated with solenoid operated valves. Additional clarification is also needed to interpret the information provided in Table 3.3-3 (e. g., MOV leak rates and its failure rates, significant cracks, if any, in pipes and thick-walled vessels and tanks).

D. Issues involved in system modeling

1. The TPP system notebook for various systems has information on trips of that particular system (e. g., the ICW system). Provide additional detail (including system response diagrams or event sequence diagrams) on the process used to model the many ways of tripping the system, the required operator recovery action, and the implementation of the TPP-specific limiting conditions of

operations (LCO) in each fault tree and the method of linking this information with a particular transient (initiating event) other than support systems.

2. Discuss the process used to treat certain system failure modes and their impact on that system and/or other systems (e.g., plant trip system), if any. Include reverse flow through a check valve, failure of reverse current relays, failure of diodes, and combination of pipe breaks of significant flow (both water and steam) diversion and failure of the associated isolation capability and impact on system unavailability.

E. <u>Issues involved in success criteria</u>

1. Provide a discussion related to the period during which the TPP facility is or will be operated with the closed block valves of the PORVs, associated LCO, if any, and its impact on frequency estimates for decay heat removal sequences and fail-to-scram sequences.

2. Provide a discussion related to the needed operator actions to keep the reactor at hot standby following a transient or a steam generator tube rupture (SGTR) event (if RHR shutdown cooling is not available). Also provide a discussion related to operator actions needed to provide coolant makeup to the Refueling Water Storage Tank (RWST) and the Auxiliary Water Storage Tank (AFWST) for extended period.

BACK-END

1. Accident initiating events were grouped according to their impact on the plant. This suggests that each group of initiating events results in plant responses which are sufficiently similar for all the initiating events included in that group. Plant responses to a particular initiating event include containment and accident progression responses in addition to the response of plant systems designed to prevent or mitigate core damage. Please describe any feedback from your containment and/or accident progression analyses in categorizing the initiating events into groups according to their plant responses.

2. In discussing "success criteria" on page 3.0-18, it is mentioned that ". . . several Level 1 questions were addressed through use of the former MAAP capability . . . ". Please list these questions and provide any assumptions made.

3. Please provide a concise discussion of the effects of severe environment on essential equipment during a severe accident scenario. Have you identified any essential equipment which would fail as a result of severe environmental effects?

4. Please augment the description of the containment purge system on page 4.0-6. Please provide a schematic diagram of the purge system which indicates the purge line penetration diameter and containment isolation valve diameter. The description provided indicates that the purge system may be used during reactor operation. Please identify the valve seat design and materials and their capability to withstand maximum severe accident containment pressure and temperature. Emergency procedures are identified which would also enable the use of the purge system for post accident containment venting. Please describe the design attributes and/or test results which provide assurance that the purge system isolation valves may be closed against post accident containment pressure and through line flow, and provide the failure probability value(s) used in the containment isolation system fault trees.

5. The IPE uses a containment isolation failure probability of 1×10^{-3}. Please provide the containment isolation system fault trees which were utilized to determine the above failure rate. Have you considered the overall effect of one order of magnitude increase in isolation system failure rate on the percentage of early containment failures? Please provide a concise discussion related to this consideration.

6. Please identify the elastomer seals which form a part of the containment boundary (i.e., equipment hatch seals, personnel access hatch seals, electrical penetration potting, etc.) Please provide a concise discussion of the design of the penetrations, seals and material properties and provide your basis for concluding that they will withstand the post accident environment (i.e., that containment structural failure will occur before elastomer seal failures resulting in large leakage areas). Your assessment should address the temperature and pressure effects of DCH and hydrogen combustion.

7. Is the value used for PRALPHAL probability (page 4.0-126) used in the CFE logic tree consistent with Surry and Seabrook PRA? Provide the conditional probability and discuss the rationale if it is not consistent with what was used in the Surry and Seabrook PRAs.

8. Please discuss how the uncertainty of the mass of RPV relocatable steel is treated in the IPE. Have you considered BNL's recommendation of an additional sensitivity run which augments core plate mass with the steel contained in the upper core guide, control rod guides, and upper and lower fuel tie plates? Please provide a concise discussion related to this issue.

9. Page 4.0-44 states that two failure modes have been evaluated using similar reference containments. Please provide the plants used in reference.

10. The containment ultimate strength pressure is given Pult= 152.8 psi; however, no indication is given with clarification of the Pult for the dry and wet scenarios. Please provide a concise clarification.

11. On page 4.0-101 the baseline scenario is evaluated for an existing analysis. Please briefly describe this analysis.

12. On page 4.0-115 how did you use the NUREG-1150 data for quantifying the CET? What percentage of plant specific and generic data were used in the CET quantification?

13. Have you performed sensitivity studies for different heat transfer coefficients between circulating debris and frozen crust? Did the differences impact any IPE conclusions?

14. Please explain in more detail the sensitivity studies you have performed in relation to the hydrogen ignition and burning, and the MAAP model parameters you have used.

    Have you performed an analysis using plant-specific information on containment subcompartment configurations to address the likelihood of local hydrogen detonation and the effects of a local hydrogen detonation on containment integrity and equipment survivability? Please provide a concise discussion as to how the local detonation issue is addressed.

15. Does modelling assumptions regarding reactor coolant system natural circulation significantly impact IPE findings, i.e., did the licensee employ natural circulation to preclude Direct Containment Heating (DCH)? If so, discuss the analytic process employed, any sensitivity studies performed and whether the supporting analysis was properly documented?

16. With respect to page 4.0-132-134 and Table 4.6-30, please provide clarification for the PDS identification, e.g. PDS III C vs. II C, PDS II H vs. I H.

    Some mentioned dominant release modes seem to contain mistakes (e.g. for PDS II C, release mode A1 is 0.0 according to Table 4.6-30). Please provide clarification related to the weighted fraction for Classes CFE, NOCCI and CFE, CCI in Table 4.6-30.

17. Paragraph 4.7.2 states that the spray removal fraction has been estimated for an appropriate PDS. How did you use the available information from NUREG/CR-4881 (mentioned but not referenced) and NUREG/CR-4551 to supplement the calculated spray removal fraction for estimating the spray removal fraction for other PDS?

18. In paragraphs 4.7.2 and 4.7.3, the fission product release fractions have been calculated using an approach similar to what were used in NUREG/CR-4881 and NUREG/CR-1150. Also the input parameters for release into the containment are derived from these NUREGs. The expert elicitation used for the second draft of the NUREG-1150 has improved the knowledge on the source terms, and some proposed radionuclide releases into the containment are different from that mentioned in NUREG/CR-4881. Have you updated these data using the final version of NUREG-1150?

19. Please provide a discussion of any plant-specific or unit-specific back-end safety features believed to be important to the Turkey Point 3 and 4 units.

20. Generic Letter 88-20 states the staff's expectation that the licensee will examine the "back-end" analysis to determine the leading contributors to unusually poor containment performance and identify any proposed plant improvements. Not withstanding the larger uncertainties in the "back-end" analysis, please discuss the process used to examine the Level 2 results to identify the leading contributors to unusually poor containment performance (UPCP). In the absence of any other definition of UPCP, early large containment release may be assumed to define UPCP.

HUMAN

1. Please discuss the human and hardware failures in recovery actions (e.g. RU3DT1D4-1) and the quantification process.

2. For those estimates whose source is 5 (guess) in Tables 3.5-2 and 3.5-3, please provide the rationale (or difficulty in developing a rationale).

3. The probability of non-recovery for event U3OPEOPE3 is very low. The reason provided in Note 3 (page 3.0-218) is "extensive simulator training in these actions." Please provide additional information about the simulator training of operators in these actions.

4. How were operator actions taken into consideration in the internal flood analysis?

5. Please provide more detail about the contribution of common cause failures, human actions, etc., to core damage frequency. What are the most important common cause failure events and human actions? Please briefly describe those crucial human actions that are unique for each unit.

   What is the contribution of human error to core damage frequency?

6. The submittal lists screening values for most human actions and indicates the final values are contained in Reference 3.4-25. Please provide two or three pages of Appendix E of Reference 3.4-25 which lists these final values.

   The IPE submittal indicates only screening values were used for human error probabilities, even in the final quantification. Why were the screening values for the important human errors not refined to provide a more realistic human error probability for these events in the final quantification?

   Please provide an example or two of how Figures 3.4-1, 3.4-2, and 3.4-3 were used in the calculation of human error probabilities.

7. In discussing PRA plant walkdowns on page 2.0-12, it is mentioned that "... often, a question raised by one of the PRA tasks could only be answered by a walkdown...". An example of these questions is also reported. Please provide a concise discussion of additional examples of important questions that were answered through plant walkdowns.

   Please briefly discuss how human factor considerations and system interaction effects were incorporated in the walkdowns performed. Your discussion should include the walkdowns performed for internal flooding.

8. On page 2.0-7 the submittal states "a methods review was performed by ERIN Corporation as a demonstration of the EPRI IPE Review Methodology Project (RP-3000-46)." Please provide a brief description of the objectives of this project, its research results, and references to any reports prepared for this project. A brief overview of the findings of this review are provided in Section 6 of the submittal. What generic insights were learned from the trial application of this project to Turkey Point?

9. Please provide a list of all performance shaping factors (PFSs) considered for the IPE and the rationale used to determine the final PSFs.

10. Please provide a copy of Reference 3.4-22.

11. On page 3.0-197 it states that Table 3.4-1 contains median probabilities. Were mean values or median values used in the quantification process?

12. Page 16 of the HRA Interim Report contains the steps used in the estimation of the human error probabilities (HEPs). Please provide information about the following related to the steps:

    (i) What is the source of the median estimates associated with step 1?

    (ii) Provide the basis for Tables 2 and 3 on page 19 associated with steps 2 and 3.

13. Were any plant-specific HEPs used in the IPE? If they were, please provide information about them and how they were derived. Otherwise, please discuss the process used to treat plant-specific human errors in the IPE.