February 1, 2001

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

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE TURKEY POINT, UNITS 3 AND 4, LICENSE RENEWAL APPLICATION

Dear Mr. Plunkett:

By letter dated September 11, 2000, Florida Power and Light (FPL), submitted for the Nuclear Regulatory Commission's (NRC) review an application pursuant to 10 CFR Part 54, to renew the operating license for Turkey Point Nuclear Plant, Units 3 and 4. The NRC staff is reviewing the information contained in the license renewal application and has identified, in the enclosure, areas where additional information is needed to complete its safety review. Specifically, the enclosed questions relate to aging management programs.

Please provide a schedule by letter, electronic mail, or telephonically for the submittal of your responses within 30 days of the receipt of this letter. Additionally, the staff would be willing to meet with FPL prior to the submittal of the responses to provide clarifications of the staff's requests for additional information.

Sincerely,

/***RA***/

Stephen S. Koenick, Project Manager
License Renewal and Standardization Branch
Division of Regulatory Improvement Program
Office of Nuclear Reactor Regulation

Docket Nos. 50-250 and 50-251

Enclosure: Request for Additional Information

cc w/encl: See next page

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E. Hylton

E-MAIL:
PUBLIC
liz_thompson@fpl.com
R. Zimmerman
J. Johnson
D. Matthews
S. Newberry
C. Grimes
C. Carpenter
B. Zalcman
J. Strosnider
R. Wessman
G. Bagchi
K. Manoly
W. Bateman
J. Calvo
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A. Murphy
W. McDowell
S. Droggitis
S. Duraiswamy
RLSB Staff

S. Hom
K. Jabbour
R. Correia
C. Julian (RII)
C. Casto (RII)
L. Wert (RII)
K. Parczewski
J. Davis
A. Hiser
A. Keim
C. Lauron
S. Hou
J. Medoff
A. Lee
C. Khan

REQUEST FOR ADDITIONAL INFORMATION
TURKEY POINT UNITS 3 AND 4
DOCKET NOS. 50-250, 50-251

AGING MANAGEMENT PROGRAMS

Section 3.8.5 - Galvanic Corrosion Susceptibility Inspection Program (LRA Section 3.1.5 of Appendix B)

RAI 3.8.5-1: Describe the operating experience involving galvanic corrosion for Turkey Point Units 3 and 4, as it relates to the industry in general.

Section 3.8.6 - Reactor Vessel Internals Inspection Program (LRA Section 3.1.6 of Appendix B)

RAI 3.8.6-1: The application describes on-going industry efforts aimed at characterizing the aging effects associated with the reactor vessel internals. What industry programs are FPL participating in to provide direction for inspection of reactor vessel internals? How will FPL integrate the results of the industry programs into the Reactor Vessel Internals Inspection Program?

RAI 3.8.6-2: Since stress corrosion cracks tend to be very tight, and the surfaces on which the cracking can occur may be rough, as-wrought or as-welded surfaces, what steps will be taken in the selection of examination technique, and what performance demonstration(s) will be used, to ensure that the features of interest (morphology and size) will be detectable with the visual examination proposed?

RAI 3.8.6-3: Timing of the reactor vessel internals inspections is important. Indicate generally when these inspections will occur (e.g., early in the renewed license period, between years 5 and 15 of the renewed license period, prior to the end of the renewed license period, etc.), and provide the basis for the selection of this timing as optimum to meet the purposes of this inspection program.

RAI 3.8.6-4: When will FPL provide for NRC staff review the specific details on this program, including the components to be inspected, requirements for detection and sizing of cracks, and acceptance criteria? The proposed FSAR supplement on this aging management program (Section 16.1.6) should be revised to clarify the intent of FPL in providing the NRC staff with these programmatic details prior to implementation of the program.

3.8.7 - Small Bore Class 1 Piping Inspection (LRA Section 3.1.7 of Appendix B)

RAI 3.8.7-1: The description states that this inspection program “will be a one-time inspection of a sample of Class 1 piping less than 4 inches in diameter.” How will the specific sample set for the inspection be determined, including which lines and which unit are to be inspected, and what measures will be taken to ensure that the sample set encompass both the range of pipe sizes less than 4 inches in diameter, and the variety of configurations (pipe, fittings, and branch connections) in the units?

<u>RAI 3.8.7-2:</u> The application indicates that this inspection will occur prior to the end of the initial operating license terms for the two units. What is the earliest point in the initial operating license term that this inspection will occur? Provide the basis for the selection of this timing as optimum to meet the purposes of this inspection program.

<u>RAI 3.8.7-3:</u> The description of this program indicates that the “volumetric [examination] technique chosen will permit detection and sizing of significant cracking of small bore Class 1 piping.” What criteria will be used to determine the smallest magnitude of “significant cracking”?

Section 3.9.1.1 - <u>ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program (LRA Section 3.2.1.1 of Appendix B)</u>

<u>RAI 3.9.1.1-1:</u> Provide a description of the programs to be used for augmented examinations. Specifically address those examinations for which commitments have been made, and those that are in addition to the ASME Code, Section XI, ISI requirements. Identify the system, components, and inspections for which credit is being taken in this AMP.

Section 3.9.2 - <u>Boraflex Surveillance Program (LRA Section 3.2.2 of Appendix B)</u>

<u>RAI 3.9.2-1:</u> Provide further information for each of the 10 elements to include a discussion of the current program and the manner in which this program is enhanced to ensure that the aging effects of Boraflex gap formation and dissolution are managed.

<u>RAI 3.9.2-2:</u> Based on the known mechanism governing the boraflex polymer matrix breakdown, boraflex degradation can be limited by minimizing disturbances to the spent fuel pool and maintaining silica equilibrium between the Boraflex panel and the surrounding water. Provide a description of the steps taken, if any, to limit the disturbance of the quiescent state of the spent fuel pool.

<u>RAI 3.9.2-3:</u> The staff agrees that blackness testing will provide information regarding gap formation consistent with the description of the change in material properties, due to irradiation, given in Section 3.6.2.2.2 of the LRA. However, justify the exclusion of the change in material properties due to both irradiation and convective forces in the spent fuel pool; i.e., a change in material properties due to dissolution of the boraflex panel and provide more detail discussing how the enhanced Boraflex Surveillance Program will determine the amount of degradation of the Boraflex material through this mechanism.

<u>RAI 3.9.2-4:</u> The applicant commits to checking the density of the panels (or other approved methods) to ascertain the physical loss of boron carbide. Provide additional details describing the nature of this commitment. The description should include what alternatives will be in place in the event that the degree to which this valid aging effect is occurring cannot be determined.

<u>RAI 3.9.2-5:</u> Blackness testing is an appropriate method for determining gap formation in the panels but is not indicative of the concentration of boron carbide remaining in the panel. Discuss how the enhanced Boraflex Surveillance Program will support conclusions drawn from the applicant’s operating experience.

RAI 3.9.2-6: The staff notes that the only aging effect discussed in Section 3.6.2.2.2 of the LRA is gap formation. Clarify how this aging effect will be detected through Blackness Testing.

RAI 3.9.2-7: Clarify how shrinkage, gap formation, and density changes of the Boraflex panels are currently trended and analyzed and provide details of how the enhanced program will affect the current analyses of these parameters.

RAI 3.9.2-8: The applicant states that the acceptability of Boraflex degradation is controlled by the assumptions in the criticality analysis. Provide details regarding how the surveillance results assure that the 5% subcriticality margin will be maintained given that dissolution of the Boraflex is not addressed in the existing program.

3.9.3 - Boric Acid Wastage Surveillance Program (LRA Section 3.2.3 of Appendix B)

RAI 3.9.3-1: Provide further detail regarding the enhancement of this program. Specifically, provide details discussing how the systems outside containment, currently inspected under other existing programs, will continue to be inspected under the enhanced Boric Acid Wastage Surveillance Program.

RAI 3.9.2-2: Discuss the exclusion of components constructed from aluminum, brass, bronze, carbon, and galvanized steel which may also be exposed to the corrosive boric acid environment.

RAI 3.9.2-3: In the case of electrical cables or insulated piping, discoloration of the insulation is used to indicate boric acid coolant leakage. Provide the acceptance criteria and the bases for this method. In addition, provide operating experience that identifies aging prior to loss of function.

RAI 3.9.3-4: Provide details regarding the evaluation of a boric acid leakage discovery to include, but not limited to, specific evaluation criteria and the bases for such criteria.

Section 3.9.4 - Chemistry Control Program (LRA Section Appendix B 3.2.4)

RAI 3.9.4 -1: Identify guidelines and/or standards including revision numbers to which the Chemistry Control Program is implemented (i.e., EPRI reports TR-105714 and TR-102134, respectively). If deviations from the guidelines, then justify the differences. If alternate means of controlling water chemistry are utilized, describe major controlling parameters, their ranges, corresponding acceptance criteria and any corrective measures which have to be taken when these criteria are exceeded.

RAI 3.9.4 - 2: Describe the Chemistry Control Program as it relates to emergency diesel fuel oil. The description should include the actions taken to prevent ingress of water into the fuel oil system. Reference any relevant standards.

RAI 3.9.4 - 3: In the discussion of “Parameters Monitored or Inspected,” the applicant specifies chemicals and water content as the parameters monitored. For microbiologically influenced corrosion (MIC), which is grouped under the aging effect of loss of material, in Appendix C, the

applicant states for the purpose of aging management review, loss of material due to MIC is not considered significant at temperatures greater than 120°F or pH greater than 10. Given these parameters, provide a discussion of how the Chemistry Control Program, which does not appear to focus on these parameters, would adequately manage this aging effect.

RAI 3.9.4 - 4: In the discussion on “Detection of Aging Effects,” the applicant states the following aging mechanisms can be minimized or prevented by the Chemistry Control Program include general corrosion, pitting corrosion, crevice corrosion, microbiologically influenced corrosion, graphitic corrosion, stress corrosion cracking, intergranular attack, corrosion fouling, and fouling caused by microbiologically influenced corrosion. These mechanisms were grouped by the applicant into the following aging effects of concern (i.e., loss of material, cracking, and fouling). However, high concentrations of impurities at crevices and locations of stagnant flow conditions could cause localized loss of material by some of the identified aging mechanisms. Provide a discussion on verification of the effectiveness of the chemistry control program (e.g., use of a one-time inspection of select components and susceptible locations) to ensure that this aging effect is not occurring.

3.9.9 - Flow-Accelerated Corrosion Program (LRA Section 3.2.9 of Appendix B)

RAI 3.9.9-1: Describe in detail the flow accelerated corrosion (FAC) program in the Turkey Point plant. Specifically, provide the following information:

- List guidance and recommendations used in developing the program.

- Specify the methodology or methodologies used for predicting loss of materials from the components subjected to FAC. If a generic methodology (e.g. CHECWORKS program developed by EPRI) is used, provide the reference. However, if it is a plant-specific methodology developed by the applicant, describe the methodology in detail.

- What are the acceptance criteria for the maximum acceptable wall thinning in the components subjected to FAC? Specify these criteria and the codes upon which they are based.

RAI 3.9.9-2: The description of the scope of the program mentioned “limited baseline inspection.” Describe the nature of this inspection.

RAI 3.9.9-3: Susceptibility to FAC can be reduced by maintaining proper water chemistry. Describe how the secondary water chemistry (treat water-secondary) will be controlled in order to achieve optimum environment for the components subjected to FAC. List any relevant guidelines or standards used to achieve this goal.

RAI 3.9.9-4: In the description of monitoring and trending activities in the program, it was indicated that in steam traps, in addition to material loss from the internal walls of piping due to FAC, material loss also occurred from the external walls due general corrosion. Both these material losses are measured by a volumetric examination performed on these lines. Explain how the loss of material from internal surfaces and from external surfaces can be determined by volumetric measurements performed on these lines when the volumetric examination

technique can only give total material losses from the piping, equal to a sum of losses from internal and external surfaces.

RAI 3.9.9-5: Describe the inspection program for the components subjected to FAC. The description should include the following:

- State methodology for selecting the components to be examined during a given outage.

- State the frequency of examination of individual components.

- Describe the techniques used for performing these examinations. i.e. ultrasonic, radiography, or visual examination. If ultrasonic examination is used, how is the wall thickness determined from the individual instrument readings.

RAI 3.9.9-6: Were the replacements for the components damaged by FAC made using the same material or in some cases was a more FAC resistant material used? If change in material is used, explain how the FAC program is impacted.

RAI 3.9.9-7: In the attribute, “Operating Experience and Demonstration,” the applicant stated that wall thinning problems have occurred. Provide more information on the operating experience related to the wall thinning observed in the components located in the main steam and turbine generators and feedwater and blowdown systems. Specifically:

- How many components experienced wall thinning beyond the acceptable level and needed replacement?

- Were there any leaks or pipe breaks in the components damaged by FAC? If such events have occurred describe them in detail.

3.9.11 - Periodic Surveillance and Preventive Maintenance Program (LRA Section 3.2.11 of Appendix B)

RAI 3.9.11-1: In page B-67, yard structures are listed as one category of structures whose aging effects are managed by the Periodic Surveillance and Preventive Maintenance Program. However, this program was not included in the last column of Table 3.6-20 which identifies specific programs and activities for aging management of yard structures. Explain this discrepancy, or make appropriate modifications either to Table 3.6-20 or in the scope of the Periodic Surveillance and Preventive Maintenance Program.

RAI 3.9.11-2: As indicated in the scope description, the Periodic Surveillance and Preventive Maintenance Program is credited for managing several aging effects including embrittlement of structures, systems, and components. However, the embrittlement effect to be managed by this program is not shown in tables related to Sections 3.3, 3.4 and 3.6.In addition, given that aging effects are detected by visual inspections, provide acceptance criteria on how embrittlement effects are managed and detected.

RAI 3.9.11-3: The submittal indicated that this program will be enhanced to address the scope of specific inspections and their documentation. As indicated in Section 16.2.11 of the updated FSAR Supplement in Appendix A, specific enhancements to the scope and documentation of some inspections performed under this program will be implemented prior to the end of the initial operating license terms for Turkey Point Units 3 and 4. Provide a description of the program enhancements sufficient to satisfy 10 CFR 54.21(a)(3).

RAI 3.9.11-4: Provide information to clarify the following:

- Since this is an existing program, describe how frequently the inspections were conducted. In addition, identify specific frequencies of component replacement.

- Describe acceptance criteria and guidelines, and identify documentation on implementation procedures for the inspections, refurbishments, and replacements.

- Show evidence regarding effectiveness of the program in the Operating Experience and Demonstration summary.

3.9.12 - Reactor Vessel Head Alloy 600 Penetration Inspection Program (LRA Section 3.2.12 of Appendix B)

RAI 3.9.12-1: NEI's integrated program for evaluating Alloy 600 VHPs in U.S. PWRs is based on the industry's generic and plant-specific responses to GL 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," and ranks the susceptibility of Alloy 600 VHPs to develop PWSCC based on probabilistic cracking models. The criteria for ranking the VHPs in the industry are based on establishing a benchmark probability that the control rod drive mechanism (CRDM) nozzles for a given facility would be equal to (normalized) the probability that a 75-percent throughwall crack would be detected and exist in the most PWSCC-degraded CRDM nozzle of the D.C. Cook Unit 2 facility relative to the time of the inspection of the VHPs at D.C. Cook Unit 2 facility in 1994. NEI's integrated program then ranks the CRDM nozzles according to the time that the benchmark probability of the nozzles for a given facility would be achieved relative to January 1, 1997. NEI normalized the CRDM nozzles in the U.S. industry into those predicted to achieve this probability within 5 years of January 1, 1997 (e.g., plants with nozzles that are considered to be highly susceptible to PWSCC - Tier 1 VHPs), those predicted to achieve this probability within 5-to-10 years of January 1, 1997 (e.g., plants with nozzles that are considered to be moderately susceptible to PWSCC - Tier 2 VHPs), and those predicted to achieve this probability within 15 or more years of January 1, 1997 (e.g., plants with nozzles that are considered to have a low susceptible to PWSCC - Tier 3 VHPs).

In its review of the NEI submittal of December 11, 1998, Turkey Point "Responses to the NRC Requests for Additional Information on Generic Letter 97-01," the NRC staff determined that the VHPs at Turkey Point Unit 4 were ranked as Tier 2 VHPs, and that the VHPs at Turkey Point Unit 3 were ranked as Tier 3 VHPs. Although the VHPs in the Turkey Point units were not selected as being those ranked and chosen for performing the integrated program's initial voluntary, volumetric inspections, FPL has modified the Alloy 600 program for the Turkey Point VHPs by committing to perform volumetric examinations of the VHPs in the Turkey Point Unit 4

RPV head. However, in Section 3.2.12, FPL did not identify if the normalized probability of cracking for the VHPs in the Turkey Point Unit 3 RPV head would achieve the equivalent ranking relative to the worst case nozzle at D.C. Cook Unit 2 within the proposed period of extended operation for the unit, and similarly did not identify when the normalized probability of cracking for the VHPs in the Turkey Point Unit 4 RPV head would achieve the equivalent ranking relative within the Tier 2 timeframe (i.e., within 2002 to 2012). Therefore with respect to the program as described in Section 3.2.12 of the TP LRA, FPL needs to:

- Respond whether the VHPs of Turkey Point Unit 3 are predicted to achieve the same probability for having a 75% throughwall, PWSCC type flaw as those detected at D.C. Cook Unit 2 within the extended operating term for the unit. If the VHPs of Turkey Point Unit 3 are predicted to achieve the same probability for having a 75% throughwall, PWSCC type flaw as those detected at D.C. Cook Unit 2 within the extended operating term for the unit, state whether FPL intends to inspect the VHPs of Turkey Point Unit 3 before or during the extended operating term for the unit. If the VHPs of Turkey Point Unit 3 are predicted to achieve the same probability for having a 75% throughwall, PWSCC type flaw as those detected at D.C. Cook Unit 2 within the extended operating term for the unit, and FPL does not intend to commit to performing voluntary volumetric examinations of these VHPs, provide a technical basis for not examining them.

- Respond when the VHPs of Turkey Point Unit 4 are predicted to achieve the same probability for having a 75% throughwall, PWSCC type flaw as those detected at D.C. Cook Unit 2 and when the planned volumetric examinations of the VHPs at Turkey Point Unit 4 are expected to take place relative to this timeframe.

3.9.13 Reactor Vessel Integrity Program (LRA Section 3.2.13 of Appendix B)

3.9.13.1 - Reactor Vessel Surveillance Capsule Removal and Evaluation (LRA Section 3.2.13.1 of Appendix B)

RAI 3.9.13-1: Table 4.4-2 in Appendix A of the LRA provides the surveillance capsule withdrawal schedule for Turkey Point Units 3 and 4. In order to monitor changes in the reactor vessel material due to neutron irradiation during the license extension period, the current reactor vessel surveillance program, which was designed based on a 40-year license, must be modified to accommodate a 60-year license. Discuss how the reactor vessel surveillance program complies with the following criteria:

- The surveillance program must provide data at neutron fluences equal to or greater than the projected peak neutron fluence at the end of the period of extended operation.

- If the last capsule is withdrawn before the 55th year, the applicant must establish reactor vessel neutron environment conditions (fluence, spectrum, temperature, and neutron flux) applicable to the surveillance data and the pressure-temperature curves. If the plant operates outside the limits established

by these conditions, the applicant must inform the NRC and determine the impact of the condition on RPV integrity.

- If the last capsule is withdrawn before the 55th year, the applicant must install neutron dosimetry to permit tracking of the fluence to the RPV.

3.9.13.2 - Fluence and Uncertainty Calculations (LRA Section 3.2.13.2 of Appendix B)

RAI 3.9.13.2-1: In Section 3.2.13.2, the applicant states that the pressure vessel fluence values are calculated in compliance with the requirements of draft guide (DG)-1053. The applicant also states that the calculations are verified using dosimetry results from the reactor vessel surveillance capsule removal and evaluation subprogram. Provide the database, the data processing (including computer codes) and the associated calculations which demonstrate adherence to the requirements of DG-1053.

3.9.14 - Steam Generator Integrity Program (LRA Section 3.2.14 of Appendix B)

RAI 3.9.14-1: It is indicated in the scope of the LRA, that this AMP applies to steam generator secondary-side integrity inspections in addition to the inspection of tubes and plugs.

- Identify the steam generator internal components that are included in the program.

- Briefly describe the examinations performed on these internal components and identify whether they are examined in accordance with the program guidelines given in NEI 97-06 (Steam Generator Program Guidelines). If they are not examined in accordance with NEI 97-06, briefly describe how the examinations differ from those specified in NEI 97-06.

RAI 3.9.14-2: Tthe steam generator integrity program is structured to meet NEI 97-06 and the plant's technical specifications. NEI 97-06 provides, among other items, guidance on the inspection and assessment of steam generator tube sleeves. Steam generator tube sleeves were not identified by the applicant in the scope of this AMP. Discuss why tube sleeves were not identified.

RAI 3.9.14-3: The submittal indicated that volumetric inspection techniques detect flaw size and depth, or alternatively, remaining sound wall thickness. No discussion is provided on the testing technique (e.g., eddy current testing) primarily utilized or the type of probes used for detecting different kinds of tube and plug degradation. Also, eddy current testing has been used in the industry to detect degradation of other internal components and the presence of loose parts. Provide a discussion on the above items as applied to tubes, plugs, internals and loose parts at Turkey Point. Indicate the standards and criteria to which these inspection techniques and personnel are qualified. Describe the inspection scope (location and probe types) used at Turkey Point.

RAI 3.9.14-4: The submittal indicated that the acceptance criteria for identified primary-to-secondary operational leakage is compared with the limits allowed by the technical specifications. However, it is also stated that the steam generator integrity program is structured to meet NEI 97-06 which requires a lower operational leakage limit than that required

by the Turkey Point technical specifications. Clarify which operational leakage limit is followed by the applicant. If the NEI 97-06 leakage limit is not followed, explain this deviation based on the applicant's stated intent to meet NEI 97-06, and the industry's determination that a lower leakage limit is more appropriate given industry experience.

RAI 3.9.14-5: Clarify how the confirmation process ensures that preventive actions are adequate and that appropriate corrective actions have been completed and are effective.

3.9.16 - Thimble Tube Inspection Program (LRA Section 3.2.16 of Appendix B)

RAI 3.9.16-1: The submittal indicated that the Thimble Tube Inspection Program is an existing program which consists of conducting an eddy current test inspection on one thimble tube (#N-05 in Unit 3) in accordance with plant procedures. Identify documentation and provide a description of the plant procedures related to thimble tube inspection. In addition, according to Section 16.2.16 of the updated FSAR Supplement in Appendix A, the Thimble Tube Inspection Program currently requires only an one-time inspection on a single tube (#N-05 in Unit 3) prior to the end of the initial operating license term for Turkey Point Unit 3, and the data of this inspection will be evaluated to determine the need for additional inspections. Due to potential uncertainties in wear rate, provide justification of the adequacy of a single tube inspection. In addition, provide criteria that will be used to determine the scope of additional tests, if necessary.

RAI 3.9.16-2: Can a thimble tube be isolated from coolant leak? Describe the corrective actions mentioned in page B-88 if a tube leak does occur.

TIME-LIMITED AGING ANALYSES

4.2 - Reactor Vessel Irradiation Embrittlement

4.2.1 Pressurized Thermal Shock

RAI 4.2.1-1: Section 4.2.1 of the LRA provides the calculated RT_{PTS} values at 48 effective full power years (EFPY) for Turkey Point Units 3 and 4. The RT_{PTS} value for the circumferential welds in both units is 297.4°F. The LRA did not provide a) the 48 EFPY fluence, b) the weld chemistry, or c) the analysis in accordance with 10 CFR 50.61 (c) (1) and (2) that resulted in the RT_{PTS} value. Provide items a) through c) and the impact of the Charpy data from the integrated Turkey Point Units 3 and 4 surveillance program on the assessment. Include a comparison of the chemistry factor calculated from the 10 CFR 50.61 Tables to the Chemistry Factor calculated from surveillance data and the appropriate Margin terms in order to demonstrate that the RT_{PTS} value is conservative.

4.2.2 Upper Shelf Energy

RAI 4.2.2-1: In section 4.2.2 of the LRA, the applicant cites reference 4.2-4, "BAW-2312, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of Turkey Point Units 3 and 4 for Extended Life Through 48 Effective Full Power Years, B&W, November 1997" as a basis for extending their upper-shelf energy (USE) equivalent margins analysis (EMA) into the period of extended operation. The applicant also stated that Appendix K of ASME Section XI

was used to demonstrate a continued, acceptable EMA. The staff was unable to find BAW-2312 document on the NRC docket. Since the LRA does not give sufficient detail of how the EMA was extended, provide BAW-2312, and a summary of the methodology used to extend the applicability of the EMA. In addition, evaluate the impact of the Charpy data from the integrated (Turkey Point Units 3 and 4) surveillance program on the assessment.

4.3.2 - Reactor Vessel Underclad Cracking

RAI 4.3.2-1: Section 4.3.2 of the Turkey Point LRA, indicates that a generic evaluation of underclad cracks had been extended to 60 years using fracture mechanics evaluations based on a representative set of design transients with the occurrences extrapolated to cover 60 years of service.

- If the evaluation has been previously submitted for staff review, identify the report and the staff safety evaluation.
- If the evaluation has not been submitted for staff review, provide the analysis.
- Compare the transients in the 60-year generic evaluation to the Turkey Point design transients and explain why the crack growth projected in the 60-year generic evaluation will bound the crack growth projected for Turkey Point in 60 years of operation.

4.7.1 - Bottom Mounted Instrumentation Thimble Tube Wear

RAI 4.7.1-1: The submittal indicated that, in response to NRC Bulletin 88-09, eddy current test inspections of thimble tubes in Units 3 and 4 were conducted in the early 90's, and tube wall wear rates were established in both units. Based on these wear rates and the time-limited aging analysis (TLAA) results, only a single tube (#N-05 in Unit 3) will require inspection for the extended operation. Identify the wear rates and describe the TLAA processes and results, including assumptions used and analysis results to justify that the acceptance criterion of 70% wall loss are met for extended operation of all thimble tubes except the tube #N-05 in Unit 3. Note that the wear rate may increase with time when flow-induced thimble tube vibrations become more severe due to increased wear. TLAA based on previous inspection results obtained in early 1990's may not be realistic without verification. Confirm that an evaluation was performed in the TLAA to ensure adequate coverage of potential uncertainties in wear rates.

Florida Power & Light Company

Turkey Point Nuclear Plant

Mr. S. Ross, Attorney
Florida Power & Light Company
P.O. Box 14000
Juno Beach, Florida 33408-0420

Mr. Robert J. Hovey, Site
Vice President
Turkey Point Nuclear Plant
Florida Power and Light Company
9760 SW. 344th Street
Florida City, Florida 33035

County Manager
Miami-Dade County
111 NW 1 Street, 29th Floor
Miami, Florida 33128

Senior Resident Inspector
Turkey Point Nuclear Plant
U.S. Nuclear Regulatory Commission
9762 SW. 344th Street
Florida City, Florida 33035

Mr. William A. Passetti, Chief
Department of Health
Bureau of Radiation Control
2020 Capital Circle, SE, Bin #C21
Tallahassee, Florida 32399-1741

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

Mr. Robert Butterworth
Attorney General
Department of Legal Affairs The Capitol
Tallahassee, Florida 32304

Mr. Donald Jernigan
Plant Manager
Turkey Point Nuclear Plant
Florida Power and Light Company
9760 SW. 344th Street
Florida City, Florida 33035

Mr. Steve Franzone
Licensing Manager
Turkey Point Nuclear Plant
Florida Power & Light Company
9760 SW. 344th Street
Florida City, Florida 33035

Mr. J.A. Stall
Vice President - Nuclear Engineering
Florida Power & Light Company
P.O. Box 14000
Juno Beach, Florida 33408-0420

Mr. Douglas J. Walter
Nuclear Energy Institute
1776 I STreet NW
Suite 400
Washington, D.C. 20006

Mr. Stephen T. Hale
Turkey Point Nuclear Point
Florida Power & Light Company
9760 S.W. 344 Street
Florida City, Florida 33035

Ms. Joette Lorion
13015 SW 90 Court
Miami, Florida 33176

Ms. E.A. Thompson
Project Manager for License Renewal
Turkey Point Nuclear Plant
Florida Power and Light Company
9760 SW 344th Street
Florida City, Florida 33035

Mr. Mark P. Oncavage
12200 S.W. 110th Avenue
Miami, Florida 33176-4520

David R. Lewis, Esq.
Shaw Pittman
2300 N Street, NW
Washington, DC 20037