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

October 4, 1999

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

SUBJECT: COMPLETION OF LICENSING ACTION FOR GENERIC LETTER 98-01 AND SUPPLEMENT 1 - TURKEY POINT PLANT, UNITS 3 AND 4 (TAC NOS. MA1900 AND MA1901)

Dear Mr. Plunkett:

On May 11, 1998, the U.S. Nuclear Regulatory Commission (NRC) issued Generic Letter (GL) 98-01, "Year 2000 Readiness of Computer Systems at Nuclear Power Plants," to all holders of operating licenses for nuclear power plants, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel. The NRC issued the GL to ensure that licensees were adequately addressing potential date-related errors with computer systems, embedded software devices, and software applications sufficiently in advance of the Year 2000 (Y2K) rollover date to ensure that nuclear power plants will be in a stable, safe condition during the Y2K transition.

Subsequent to the issuance of GL 98-01, increased public awareness and government attention to the Y2K issue resulted in concern over, not only public health and safety of nuclear power plants, but the ability of nuclear power plants to continue to provide power to the national electric power grid. Therefore, on January 14, 1999, the staff issued Supplement 1 to GL 98-01 to provide addressees with a voluntary, alternative response to that required in Item (2) of the GL. In responding to Supplement 1, addressees were asked to confirm Y2K readiness of the facility with regard to those systems within the scope of the license and NRC regulations, as well as those systems required for continued operation of the facility after January 1, 2000. Addressees were permitted to voluntarily respond to Supplement 1 to the GL on or before July 1, 1999, in lieu of Item (2) of the GL.

In response to GL 98-01, you provided letters dated August 4, 1998, June 28 and July16, 1999, for Turkey Point Plant, Units 3 and 4. These submittals provided the information requested by the GL. The staff has reviewed your responses, and has concluded that all requested information has been provided. Therefore, we consider GL 98-01 to be closed for Turkey Point Plant, Units 3 and 4.

NUREG-1706, "Year 2000 Readiness in U.S. Nuclear Power Plants," provides a status of nuclear power plant Year 2000 readiness as of September 1, 1999, and a description of NRC actions to determine Y2K readiness in operating U.S. reactors. The staff assessment of Y2K readiness consisted of independently evaluating nuclear power plant licensee Y2K readiness program processes, reviewing licensee responses to NRC requests for reporting Y2K

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readiness (i.e., GL 98-01 and Supplement 1), and combining the results on these assessments to achieve a high level of assurance that each facility will operate safely during the transition from 1999 to 2000 and on other Y2K sensitive dates. NUREG-1706 is linked to the NRC home page (http:/www.nrc.gov/NRC/NEWS/year2000.html).

We would like to acknowledge the efforts you have completed to date in preparing your plant for the Y2K transition and appreciate your continued cooperation on this important issue.

If you have any questions regarding this matter, please contact me at (301) 415-1496.

Sincerely,

Kahtan N. Jabbour

Kahtan N. Jabbour, Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-250 and 50-251

cc: See next page

Mr. T. F. Plunkett
Florida Power and Light Company

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readiness (i.e., GL 98-01 and Supplement 1), and combining the results on these assessments to achieve a high level of assurance that each facility will operate safely during the transition from 1999 to 2000 and on other Y2K sensitive dates. NUREG-1706 is linked to the NRC home page (http:/www.nrc.gov/NRC/NEWS/year2000.html).

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Sincerely,

Original signed by:
Kahtan N. Jabbour, Senior Project Manager, Section 2
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Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-250 and 50-251

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L-2000-010
10 CFR 50.55a

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

Re: Turkey Point Unit 3
Docket No. 50-250
Risk-Informed Inservice Inspection Program

In accordance with 10 CFR 50.55a(a)(3), Florida Power and Light Company (FPL) requests to revise the Turkey Point Unit 3 ISI Program for Class 1 piping only, through the use of the attached Risk-Informed Inservice Inspection Program (RI-ISI) as an alternative to the current requirements of the ASME Boiler and Pressure Vessel Code Section XI, 1989 Edition, as required by 10CFR50.55a.

The proposed revisions to the current ISI Program for Class 1 piping only are based on the risk-informed process described in Westinghouse Owners Group (WOG) WCAP-14572, Revision 1-NP-A, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report," and WCAP-14572, Revision 1-NP-A, Supplement 1, "Westinghouse Structural Reliability and Risk Assessment (SRRA) Model for Piping Risk-Informed Inservice Inspection."

The attached Risk-Informed Inservice Inspection Program supports the conclusion that the proposed alternative provides an acceptable level of quality and safety as required by 10 CFR 50.55a(a)(3)(i). This program also meets the intent and principles of Regulatory Guides 1.174 and 1.178. This program submittal has been reviewed by the Plant Nuclear Safety Committee.

FPL requests NRC approval of this WOG Risk-Informed Inservice Inspection Pilot Program within 6 months to support timely implementation of the Program. It should also be noted that FPL considers implementation of this RI-ISI Pilot Program to be a Cost Beneficial Licensing Action (CBLA).

Should there be any questions concerning this submittal, please contact us.

Very truly yours,

R. J. Hovey
Vice President
Turkey Point Plant

GSS

Attachment

cc: Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, Turkey Point Plant

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ATTACHMENT TO L-2000-010

FLORIDA POWER AND LIGHT

TURKEY POINT UNIT 3

Risk-Informed Inservice Inspection Program Using the Westinghouse Owners Group (WOG) Methodology (WCAP-14572, Revision 1-NP-A, February 1999)

December 1999

003681509

RISK-INFORMED INSERVICE INSPECTION PROGRAM PLAN

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1. INTRODUCTION/RELATION TO NRC REGULATORY GUIDE RG-1.174

Introduction

Inservice inspections (ISI) are currently performed on piping to the requirements of the ASME Boiler and Pressure Vessel Code Section XI, 1989 Edition as required by 10CFR50.55a. Turkey Point Unit 3 is currently in the third inspection interval as defined by the Code for Program B.

The objective of this submittal is to request a change to the ISI program plan for Class 1 piping only through the use of a risk-informed inservice inspection (RI-ISI) program. The risk-informed process used in this submittal is described in Westinghouse Owners Group WCAP-14572, Revision 1-NP-A, , "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report," and WCAP-14572, Revision 1-NP-A, Supplement 1, "Westinghouse Structural Reliability and Risk Assessment (SRRA) Model for Piping Risk-Informed Inservice Inspection," (referred to as "WCAP-14572, A-version" for the remainder of this document).

As a risk-informed application, this submittal meets the intent and principles of Regulatory Guides 1.174 and 1.178. Further information is provided in Section 3.10 relative to defense-in-depth.

PRA Quality

The Turkey Point Level 1 and Level 2 Probabilistic Safety Assessment (PSA) model, 1997 update, was used to evaluate the consequences of pipe ruptures during operation in Modes 1 and 2. The base core damage frequency (CDF) and base large, early release frequency (LERF) from this version of the PSA model are 6.09E-05/yr and 1.00E-05/yr, respectively.

The Turkey Point PSA model is an updated version of the original Turkey Point Individual Plant Examination (IPE) submittal. Prior to the IPE being submitted to the U.S. Nuclear Regulatory Commission (NRC), a peer review was conducted by an outside contractor. All review findings were addressed prior to the IPE submittal to NRC on June 25, 1991. Following the submittal, the NRC chose to apply a "Step 1" and a "Step 2" review of the IPE. The "Step 2" portion consisted of a more detailed review, including an on-site visit by an NRC review team. After resolving the findings of this review, a revised IPE was submitted to NRC in March 1992. The NRC safety evaluation report (SER) for the IPE was issued thereafter in October 1992. The SER and the associated technical report were very positive in their assessment of the Turkey Point IPE. The few comments on the submittal were minor and were addressed by Florida Power and Light (FPL) and closed out.

Since the IPE, the FPL Reliability and Risk Assessment Group (RRAG) has maintained the Turkey Point PSA model consistent with the plant configuration as it has evolved. The PSA computer models are updated on an as-needed basis for various reasons, such as plant changes and modifications, procedure changes, accrual of new plant data, modeling improvements, advances in PSA technology, and issuance of new industry PSA standards. These changes are implemented and documented in a timely manner to ensure that risk analyses performed in support of plant operation reflect the current plant configuration, operating philosophy, transient history, system and component performance.

The PSA maintenance and update process is governed by the RRAG PSA procedures. Updates to the Turkey Point PSA model are documented and reviewed via engineering calculations and evaluations in

accordance with the FPL Engineering Department's Quality Instructions and RRAG procedures. As further verification for this application, the RI-ISI evaluation included a determination that the PSA model and supporting documentation accurately reflects the current Turkey Point plant configuration.

2. PROPOSED ALTERNATIVE TO ISI PROGRAM

2.1 ASME Section XI

ASME Section XI Class 1 Categories B-F, B-J, etc. currently contain the requirements for examining (via non-destructive examination (NDE) Class 1 piping components. This current program is limited to ASME Class 1 piping, including piping currently exempt from requirements. The alternative RI-ISI program for piping is described in WCAP-14572, A-Version. The Class 1 RI-ISI program will be substituted for the current examination program on piping in accordance with 10 CFR 50.55a(a)(3)(i) by alternatively providing an acceptable level of quality and safety. Other non-related portions of the ASME Section XI Code will be unaffected. WCAP-14572, A version, provides the requirements defining the relationship between the risk-informed examination program and the remaining unaffected portions of ASME Section XI.

2.2 Augmented Programs

There are no augmented inspection programs for the Turkey Point Unit 3 Class 1 piping systems.

3. RISK-INFORMED ISI PROCESSES

The processes used to develop the RI-ISI program are consistent with the methodology described in WCAP-14572, A-Version.

The process that is being applied, involves the following steps:

- Scope Definition
- Segment Definition
- Consequence Evaluation
- Failure Assessment
- Risk Evaluation
- Expert Panel Categorization
- Element/NDE Selection
- Implement Program

Feedback Loop

There are no deviations to the process described in WCAP-14572, A-Version.

3.1 Scope of Program

The scope of this program is limited to the Class 1 piping, including piping exempt from current requirements. The Class 1 piping systems included in the risk-informed ISI program are provided in Table 3.1-1. For Turkey Point Unit 3, because of the vintage of the plant, the Class 1 piping boundaries include from the reactor coolant system up to, in most cases, the second isolation valve. This includes piping through the excess letdown and regenerative heat exchangers in the chemical and volume control system.

3.2 Segment Definitions

Once the scope of the program is determined, the piping for these systems is divided into segments.

The number of pipe segments defined for the Class 1 piping systems are summarized in Table 3.1-1. The as-operated piping and instrumentation diagrams were used to define the segments. The simplified drawings showing the piping segments are provided in the Turkey Point Engineering Evaluation PTN-ENG-SEOS-99-0152.

3.3 Consequence Evaluation

The consequences of pressure boundary failures are measured in terms of core damage and large early release frequency. The impact on these measures due to both direct and indirect effects was considered.

A review of the license basis of Turkey Point (Turkey Point Units 3 and 4 Updated Final Safety Analysis Report and supporting documents) was performed to determine the potential impact of the indirect effects of pipe leak or rupture inside containment. As a result of the review, it was concluded that the containment structure and the safety related components inside containment are adequately protected from pipe failures such that the effects of a failure are limited to direct effects.

3.4 Failure Assessment

Failure estimates were generated utilizing industry failure history, plant specific failure history and other industry relevant information.

The engineering team that performed this evaluation used the Westinghouse structural reliability and risk assessment (SRRA) software program (described in WCAP-14572, Revision 1-NP-A, Supplement 1) to aid in the process. Generally, the SRRA code was used to estimate where the possible ranges of failure probability would fall. The final probability selected was determined by the team members using the relevant information and industry experience.

Table 3.4-1 summarizes the failure probability estimates by failure mechanism and also identifies the systems susceptible to these mechanisms.

No augmented inspections are performed for the Class 1 piping.

3.5 Risk Evaluation

Each piping segment within the scope of the program was evaluated to determine its CDF and LERF due to the postulated piping failure. Calculations were also performed with and without operator action.

Once this evaluation was completed, the total pressure boundary core damage frequency and large early release frequency were calculated by summing across the segments for each system. The results of these calculations are presented in Table 3.5-1. The expected value for core damage frequency due to piping failure without operator action is 3.69E-05/year, and with operator action is 3.60E-05/year. The expected value for large early release frequency due to piping failure without operator action is 2.62E-06/year, and with operator action is 2.56E-06/year.

To assess safety significance, the risk reduction worth (RRW) and risk achievement worth (RAW) importance measures were calculated for each piping segment.

3.6 Expert Panel Categorization

The final safety determination (i.e., high and low safety significance) of each piping segment was made by the expert panel using both probabilistic and deterministic insights. The expert panel was comprised of personnel who have expertise in the following fields; probabilistic safety assessment, inservice examination, nondestructive examination, stress and material considerations, plant operations, plant and industry maintenance, repair, and failure history, system design and operation, and SRRA methods including uncertainty. Maintenance Rule Expert Panel members were used to ensure consistency with the other PSA applications.

The expert panel had the following positions represented during the expert panel meetings.

- Probabilistic Safety Assessment (PSA engineer)
- Maintenance Rule (Chairman)
- Operations (Senior Reactor Operator)
- Inservice Inspection (ISI&NDE)
- Plant & Industry Maintenance , Repair, and Failure History (System Engineer)
- Materials Engineer
- Stress Engineer

A minimum of 4 members filling the above positions constituted a quorum. This core team of panel members was supplemented by other experts, including a metallurgist and piping stress engineer, as required for the piping system under evaluation.

The expert panel chairperson was appointed by the Turkey Point Engineering Manager. The chairperson conducted the meeting.

Members received training and indoctrination in the risk-informed inservice inspection selection process. They were indoctrinated in the application of risk analysis techniques for ISI. These techniques included risk importance measures, threshold values, failure probability models, failure mode assessments, PSA modeling limitations and the use of expert judgment. Training documentation is maintained with the expert panel's records.

Worksheets were provided to the panel on each system for each piping segment, containing information pertinent to the panel's selection process. This information, in conjunction with each panel member's own expertise and other documents as appropriate, were used to determine the safety significance of each piping segment.

A consensus process was used by the expert panel. Consensus is defined as unanimous during first consideration and 2/3 (rounding conservatively) of members or alternates present in the second or subsequent considerations. The chairperson allowed appropriate time duration between considerations for deliberation.

Meeting minute records were generated. The minutes included the names of members in attendance and whether a quorum was present. The minutes contained relevant discussion summaries and the results of membership voting.

3.7 Identification of High Safety Significant Segments

The number of high safety significant segments for each system, as determined by the expert panel, is shown in Table 5-1.

3.8 Structural Element and NDE Selection

The structural elements in the high safety significant piping segments were selected for inspection and appropriate non-destructive examination methods were defined.

The initial program being submitted addresses the high safety significant (HSS) piping components placed in regions 1 and 2 of Figure 3.7-1 and described in Section 3.7.1 in WCAP-14572, A-Version. Region 3 piping components, which are low safety significant, are to be considered in an Owner Defined Program and is not considered part of the program requiring NRC approval. Region 1, 2, 3 and 4 piping components will continue to receive Code required pressure testing, as part of the current ASME Section XI program. For the 201 piping segments that were evaluated in the RI-ISI program, Region 1 contains 18 segments, Region 2 contains 12 segments, no segments are contained in Region 3, and Region 4 contains 171 segments.

The number of locations to be inspected in applicable HSS segments was determined using a Westinghouse statistical (Perdue) model as described in section 3.7 of WCAP-14572, A-Version. The 12 HSS piping segments in Region 2, including the RCS primary loop piping, were evaluated using this model. The welds in several of the HSS segments in Region 1 are socket welds where neither surface nor volumetric examinations are possible. The pressurizer surge line included thermal stratification as a primary degradation mechanism. For these segments, where the Perdue Model is not applicable, the guidance in Section 3.7.3 of WCAP-14572, A-Version was followed to identify appropriate exam locations. At this time, all 7 locations (welds) in the surge line will be examined as part of the RI-ISI program.

The segments categorized as HSS by the plant expert panel which include socket welds consist of piping with a nominal diameter of 2 inches or less. The socket welds in these segments cannot be individually examined by any currently available NDE methods that are appropriate for the degradation mechanism of intent. Therefore, for these segments a focused visual VT-2 exam will be performed during the system pressure test each refueling outage with emphasis on a specific and potentially limiting location within each segment.

Table 4.1-1 in WCAP-14752, A-Version, was used as guidance in determining the examination requirements for the HSS piping segments. VT-2 visual examinations are scheduled in accordance with the station's pressure test program which remains unaffected by the risk-informed inspection program.

Additional Examinations

Since the risk-informed inspection program will require examinations on a large number of elements constructed to lesser pre-service inspection requirements, the program in all cases will determine through an engineering evaluation the root cause of any unacceptable flaw or relevant condition found during examination. The evaluation will include the applicable service conditions and degradation mechanisms to establish that the element(s) will still perform their intended safety function during subsequent operation. Elements not meeting this requirement will be repaired or replaced.

The evaluation will include whether other elements on the segment or segments are subject to the same root cause and degradation mechanism. Additional examinations will be performed on these elements up to a number equivalent to the number of elements required to be initially inspected on the segment or segments. If unacceptable flaws or relevant conditions are again found similar to the initial problem, the remaining elements identified as susceptible will be examined. No additional examinations will be performed if there are no additional elements identified as being susceptible to the same service related root cause conditions or degradation mechanism.

3.9 Program Relief Requests

Alternate methods are specified to ensure structural integrity in cases where examination methods cannot be applied due to limitations such as inaccessibility or radiation exposure hazard.

An attempt has been made to provide a minimum of >90% coverage (per Code Case N-460) for all of the risk-informed examinations. However, some limitations will not be known until the examination is performed, since some locations will be examined for the first time by the specified techniques. In instances where a location may be found at the time of the examination that it does not meet >90%

coverage, the process outlined in Section 4.0 (Inspection Program Requirements) of WCAP-14572, A-Version will be followed.

3.10 Change in Risk

The risk-informed ISI program has been done in accordance with Regulatory Guide 1.174, and the risk from implementation of this program is expected to slightly decrease when compared to that estimated from current requirements.

A comparison between the proposed RI-ISI program and the current ASME Section XI ISI program was made to evaluate the change in risk. The approach evaluated the change in risk with the inclusion of the probability of detection as determined by the SRRA model. This evaluation resulted in the identification of 7 piping segments for which examinations will continue to be performed.

The results from the risk comparison are shown in Table 3.10-1. As seen from the table, the overall RI-ISI program slightly reduces or maintains the risk associated with piping CDF/LERF, with respect to the current Section XI program, while reducing the number of examinations. Table 3.10-1 also includes the systems that are the significant contributors to the overall risk for both the current program and the RI-ISI program. The primary basis for being able to slightly reduce (or maintain) risk with a reduced number of examinations is that exams are now being placed on piping segments that are high safety significant, and in some cases are not inspected by NDE in the current ASME Section XI ISI program.

Defense-In-Depth

The reactor coolant piping will continue to receive a system leakage test and visual VT-2 examination as currently required by the Code. Surface and volumetric examinations will also continue on the main reactor coolant piping and main safety injection lines (downstream of first check valve) as part of the RI-ISI program (segments categorized HSS). These locations, which include reactor vessel, steam generator, and pressurizer dissimilar metal welds determined by the RI-ISI program for Turkey Point Unit 3, assure that "defense-in-depth" is maintained. No additional inspection locations are required to meet "defense-in-depth."

4. IMPLEMENTATION AND MONITORING PROGRAM

Upon approval of the RI-ISI program, procedures that comply with the guidelines described in WCAP-14572, A-Version, will be prepared to implement and monitor the program. The new program will be integrated into the existing ASME Section XI interval. No changes to the Technical Specifications or the Final Safety Analysis Report are necessary for program implementation.

The applicable aspects of the Code not affected by this change would be retained, such as inspection methods, acceptance guidelines, pressure testing, corrective measures, documentation requirements, and quality control requirements. Existing ASME Section XI program implementing procedures would be retained and would be modified to address the RI-ISI process, as appropriate. Additionally, the procedures will be modified to include the high safety significant locations in the program.

The proposed monitoring and corrective action program will contain the following elements:

A. Identify

B. Characterize

C. (1) Evaluate, determine the cause and extent of the condition identified

(2) Evaluate, develop a corrective action plan or plans

D. Decide

E. Implement

F. Monitor

G. Trend

The RI-ISI program is a living program requiring feedback of new relevant information to ensure the appropriate identification of high safety significant piping locations. As a minimum, risk ranking of piping segments will be reviewed and adjusted on an ASME period basis. Significant changes may require more expedited adjustment as directed by NRC bulletin or Generic Letter requirements, or by plant specific feedback.

5. PROPOSED ISI PROGRAM PLAN CHANGE

A comparison between the RI-ISI program and the current ASME Section XI program requirements for piping is given in Table 5-1.

The plant will be performing examinations on elements not currently required to be examined by ASME Section XI. An example of these additional examinations is that several elements (segments) currently classified as exempt from examination as NPS 1 and smaller shall be included into the program plan, for Class 1 only.

The initial program will be started in the third period of the third interval. For example, the second inspection period of the third inspection interval for Unit 3 ends on February 21, 2001. Currently, 55% of the exams in the Section XI program have been performed, meeting the 50% requirement for the end of the second inspection period of the current interval. Therefore, exams initially planned during the upcoming Turkey Point Unit 3 outage in February 28, 2000 will be delayed anticipating approval and implementation of the RI-ISI program during the third period.

6.0 REFERENCES/DOCUMENTATION

WCAP-14572,Revision 1-NP-A, , "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report," February 1999

WCAP-14572, Revision 1-NP-A, Supplement 1, "Westinghouse Structural Reliability and Risk Assessment (SRRA) Model for Piping Risk-Informed Inservice inspection," February 1999

Supporting Onsite Documentation

The onsite documentation is contained within the Engineering Evaluation PTN-ENG-SEOS-99-0152 and the following calculations.

CN-RRA-99-15 Revision 0, "Turkey Point Unit 3 Risk Ranking Analysis," December 1999. (Westinghouse Proprietary)

CN-RRA-99-17 Revision 0, "Turkey Point Unit 3 Risk-Informed ISI Program – Segment Definition," December 1999. (Westinghouse Proprietary)

CN-RRA-99-35, "Turkey Point Unit 3 Risk-Informed ISI Program – Expert Panel Worksheets and MS Access Database," December 1999. (Westinghouse Proprietary)

CN-RRA-99-38 Revision 0, "Turkey Point Unit 3 Perdue Model Calculations for Risk-Informed ISI Pilot Program," December 1999. (Westinghouse Proprietary)

CN-RRA-99-40, "Turkey Point Unit 3 Revised Failure Probability Calculations with WinSRRA Version 1.1.0," December 1999. (Westinghouse Proprietary)

CN-RRA-99-41, "Turkey Point Unit 3 RI-ISI Program - Delta-Risk Calculation," December 1999. (Westinghouse Proprietary)

Table 3.1-1 System Selection and Segment Definition for Class 1 Piping			
System Description	PRA	Section XI	Number of Segments
CH - Chemical & Volume Control	Yes	Yes	70
RC - Reactor Coolant	Yes	Yes	102
SI - Safety Injection	Yes	Yes	29
Total			201

Table 3.4-1 Failure Probability Estimates (without ISI)		
Failure Mechanism	Failure Probability Range (Small Leak Probability @ 40 years, no ISI)	Susceptible Systems
Corrosion	N/A	None
Fatigue	1.0E-08 to 1.4E-04	RC, SI, CH
Stress Corrosion Cracking	N/A	None
Striping/Stratification	4.0E-04	Pressurizer surge line
Vibratory Fatigue (Low/Moderate)	1.7E-08 to 5.9E-5	RC, SI, CH
Wastage	N/A	None

Table 3.5-1

Number of Segments and Piping Risk Contribution by System (without ISI)
(Values shown are expected values)

System	# of Segments	CDF without Operator Action (/yr)	CDF with Operator Action (/yr)	LERF without Operator Action (/yr)	LERF with Operator Action (/yr)
CH	70	2.52E-05	2.44E-05	1.51E-06	1.47E-06
RC	102	1.17E-05	1.16E-05	1.10E-06	1.10E-06
SI	29	7.02E-10	7.02E-10	5.52E-11	5.52E-11
TOTAL	201	3.69E-05	3.60E-05	2.62E-06	2.56E-06

Table 3.10-1

COMPARISON OF CDF/LERF FOR CURRENT SECTION XI AND RISK-INFORMED ISI PROGRAMS AND THE SYSTEMS WHICH CONTRIBUTED SIGNIFICANTLY TO THE CHANGE

Case (Systems Contributing to Change)	Piping CDF/LERF Current Section XI	Piping CDF/LERF Risk-Informed
CDF No Operator Action (RC, CH)	1.66E-05	1.65E-05
CDF with Operator Action (RC, CH)	1.66E-05	1.65E-05
LERF No Operator Action (RC, CH)	1.02E-06	1.01E-06
LERF with Operator Action (RC, CH)	1.01E-06	1.01E-06

Note: CDF/LERF values include credit for leak detection also.

Table 5-1

STRUCTURAL ELEMENT SELECTION
RESULTS AND COMPARISON TO ASME SECTION XI
1989 EDITION REQUIREMENTS

System	Number of High Safety-Significant Segments (No. in Augmented Program)	RI-ISI Program High Safety-Significant Structural Elements[a]	ASME Section XI ISI Program 1989 Edition Examination Category Weld Selections		Total Number of Segments Credited in Augmented Programs
		CLASS 1	B-F	B-J	
CH	16 (0)	16[b]	0	57	0
RC	14 (0)	19 + 1[b] + 7[c]	18	86	0
SI	0 (0)	0	0	63	0
Total	30 (0)	19 + 17[b] + 7[c]	18	206	0

Summary: Current ASME Section XI selects a total of 224 non-destructive exam locations while the proposed RI-ISI program selects a total of 26 exam locations (43 - 17 visual exam locations), which results in an 88% reduction.

Notes for Table 5-1

a. System pressure test requirements and VT-2 visual examinations shall continue to be performed in all ASME Code Class 1, 2, and 3 systems.
b. VT-2 area exam at specific location.
c. Examinations added for change in risk considerations.

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COMPLETION OF LICENSING ACTION FOR GENERIC LETTER 98-04 "POTENTIAL FOR DEGRADATION OF THE EMERGENCY CORE COOLING SYSTEM AND THE CONTAINMENT SPRAY SYSTEM AFTER A LOSS-OF-COOLANT ACCIDENT BECAUSE OF CONSTRUCTION. ..."

Body:
PDR ADOCK 05000250 P

Docket: 05000250, Notes: N/A

Docket: 05000251, Notes: N/A

MA

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

December 9, 1999

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

SUBJECT: COMPLETION OF LICENSING ACTION FOR GENERIC LETTER 98-04, "POTENTIAL FOR DEGRADATION OF THE EMERGENCY CORE COOLING SYSTEM AND THE CONTAINMENT SPRAY SYSTEM AFTER A LOSS-OF-COOLANT ACCIDENT BECAUSE OF CONSTRUCTION AND PROTECTIVE COATING DEFICIENCIES AND FOREIGN MATERIAL IN CONTAINMENT," DATED JULY 14, 1998 - ST. LUCIE NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. MA4103 AND MA4104)

Dear Mr. Plunkett:

On July 14, 1998, the U.S. Nuclear Regulatory Commission (NRC) issued Generic Letter (GL) 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment," to all holders of operating licenses or construction permits. The NRC issued GL 98-04 to determine the status of containment coating programs.

In GL 98-04, the NRC staff specifically requested that the licensees provide information outlined below for each of their facilities.

(1) A summary description of the plant-specific program or programs implemented to ensure that Service Level 1 protective coatings used inside the containment are procured, applied, and maintained in compliance with applicable regulatory requirements and the plant-specific licensing basis for the facility. Include a discussion of how the plant-specific program meets the applicable criteria of 10 CFR Part 50, Appendix B, as well as information regarding any applicable standards, plant-specific procedures or other guidance used for (a) controlling the procurement of coatings and paints used at the facility; (b) the qualification testing of protective coatings; and (c) surface preparation, application, surveillance, and maintenance activities for protective coatings. Maintenance activities refer to rework of degraded coatings, removing degraded coatings to sound coatings, correctly preparing the surfaces, applying new coating, and verifying the quality of coatings.

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(2) Information demonstrating compliance with item (i) or item (ii).

(i) For plants with licensing-basis requirements for tracking the amount of unqualified coatings inside the containment and for assessing the impact of potential coating debris on the operation of safety-related systems, structures, and components (SSCs) during a postulated design basis loss-of-coolant accident (DB LOCA), the following information shall be provided to demonstrate compliance:

DF01

PDR ADOCK ML993480526

(a) The date and findings of the last assessment of coatings and the planned date of the next assessment of coatings.

(b) The limit for the amount of unqualified protective coatings allowed in the containment and how this limit is determined. Discuss any conservatism in the method used to determine this limit.

(c) If a commercial-grade dedication program is being used at your facility for dedicating commercial-grade coatings for Service Level 1 applications inside the containment, discuss how the program adequately qualifies a coating for Service Level 1. Identify what standards or other guidance are currently being used to dedicate containment coatings at your facility.

(ii) For plants without the above licensing-basis requirements, information shall be provided to demonstrate compliance with the requirements of 10 CFR 50.46b(5), "Long-term cooling" and the functional capability of the safety-related containment spray system (CSS) as set forth in your licensing basis. If a licensee can demonstrate this compliance without quantifying the amount of unqualified coatings, this is acceptable. The following information shall be provided:

If a commercial-grade dedication program is not being used at your facility for qualifying and dedicating commercial-grade coatings for Service Level 1 applications, provide the regulatory and safety basis for not controlling these coatings in accordance with such a program. Additionally, explain why the facility's licensing basis does not require such a program.

In response to GL 98-04, you provided a letter dated November 9, 1998, for Turkey Point Plant, Units 3 and 4. This submittal provided the information requested by GL 98-04. Clarification was provided during a phone call that took place on November 15, 1999. The staff has reviewed your response and has concluded that all requested information has been provided, therefore, we consider GL 98-04 to be closed for your facility. We thank you for your prompt and complete response.

If you have any questions regarding this matter, please contact me at (301) 415-1496.

Sincerely,

Original signed by K.Jabbour
Kahtan N. Jabbour, Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-250 and 50-251
cc: See next page
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(a) The date and findings of the last assessment of coatings and the planned date of the next assessment of coatings.

(b) The limit for the amount of unqualified protective coatings allowed in the containment and how this limit is determined. Discuss any conservatism in the method used to determine this limit.

(c) If a commercial-grade dedication program is being used at your facility for dedicating commercial-grade coatings for Service Level 1 applications inside the containment, discuss how the program adequately qualifies a coating for Service Level 1. Identify what standards or other guidance are currently being used to dedicate containment coatings at your facility.

(ii) For plants without the above licensing-basis requirements, information shall be provided to demonstrate compliance with the requirements of 10 CFR 50.46b(5), "Long-term cooling" and the functional capability of the safety-related containment spray system (CSS) as set forth in your licensing basis. If a licensee can demonstrate this compliance without quantifying the amount of unqualified coatings, this is acceptable. The following information shall be provided:

If a commercial-grade dedication program is not being used at your facility for qualifying and dedicating commercial-grade coatings for Service Level 1 applications, provide the regulatory and safety basis for not controlling these coatings in accordance with such a program. Additionally, explain why the facility's licensing basis does not require such a program.

In response to GL 98-04, you provided a letter dated November 4, 1998, for St. Lucie Nuclear Plant, Units 1 and 2. This submittal provided the information requested by GL 98-04. Clarification was provided during a phone call that took place on November 17, 1999. The staff has reviewed your response and has concluded that all requested information has been provided; therefore, we consider GL 98-04 to be closed for your facility. We thank you for your prompt and complete response.

If you have any questions regarding this matter, please contact me at (301) 415-1496.

Sincerely,

Kahtan N. Jabbour

Kahtan N. Jabbour, Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-335 and 50- 389

cc: See next page

Mr. T. F. Plunkett
Florida Power and Light Company

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

December 9, 1999

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

SUBJECT: COMPLETION OF LICENSING ACTION FOR GENERIC LETTER 98-04, "POTENTIAL FOR DEGRADATION OF THE EMERGENCY CORE COOLING SYSTEM AND THE CONTAINMENT SPRAY SYSTEM AFTER A LOSS-OF-COOLANT ACCIDENT BECAUSE OF CONSTRUCTION AND PROTECTIVE COATING DEFICIENCIES AND FOREIGN MATERIAL IN CONTAINMENT," DATED JULY 14, 1998 - TURKEY POINT PLANT, UNITS 3 AND 4 (TAC NOS. MA4111 AND MA4112)

Dear Mr. Plunkett:

On July 14, 1998, the U.S. Nuclear Regulatory Commission (NRC) issued Generic Letter (GL) 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment," to all holders of operating licenses or construction permits. The NRC issued GL 98-04 to determine the status of containment coating programs.

In GL 98-04, the NRC staff specifically requested that the licensees provide information outlined below for each of their facilities.

(1) A summary description of the plant-specific program or programs implemented to ensure that Service Level 1 protective coatings used inside the containment are procured, applied, and maintained in compliance with applicable regulatory requirements and the plant-specific licensing basis for the facility. Include a discussion of how the plant-specific program meets the applicable criteria of 10 CFR Part 50, Appendix B, as well as information regarding any applicable standards, plant-specific procedures or other guidance used for (a) controlling the procurement of coatings and paints used at the facility; (b) the qualification testing of protective coatings; and (c) surface preparation, application, surveillance, and maintenance activities for protective coatings. Maintenance activities refer to rework of degraded coatings, removing degraded coatings to sound coatings, correctly preparing the surfaces, applying new coating, and verifying the quality of coatings.

(2) Information demonstrating compliance with item (i) or item (ii).

(i) For plants with licensing-basis requirements for tracking the amount of unqualified coatings inside the containment and for assessing the impact of potential coating debris on the operation of safety-related systems, structures, and components (SSCs) during a postulated design basis loss-of-coolant accident (DB LOCA), the following information shall be provided to demonstrate compliance:

(a) The date and findings of the last assessment of coatings and the planned date of the next assessment of coatings.

(b) The limit for the amount of unqualified protective coatings allowed in the containment and how this limit is determined. Discuss any conservatism in the method used to determine this limit.

(c) If a commercial-grade dedication program is being used at your facility for dedicating commercial-grade coatings for Service Level 1 applications inside the containment, discuss how the program adequately qualifies a coating for Service Level 1. Identify what standards or other guidance are currently being used to dedicate containment coatings at your facility.

(ii) For plants without the above licensing-basis requirements, information shall be provided to demonstrate compliance with the requirements of 10 CFR 50.46b(5), "Long-term cooling" and the functional capability of the safety-related containment spray system (CSS) as set forth in your licensing basis. If a licensee can demonstrate this compliance without quantifying the amount of unqualified coatings, this is acceptable. The following information shall be provided:

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If you have any questions regarding this matter, please contact me at (301) 415-1496.

Sincerely,
Original signed by:
Kahtan N. Jabbour, Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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(a) The date and findings of the last assessment of coatings and the planned date of the next assessment of coatings.

(b) The limit for the amount of unqualified protective coatings allowed in the containment and how this limit is determined. Discuss any conservatism in the method used to determine this limit.

(c) If a commercial-grade dedication program is being used at your facility for dedicating commercial-grade coatings for Service Level 1 applications inside the containment, discuss how the program adequately qualifies a coating for Service Level 1. Identify what standards or other guidance are currently being used to dedicate containment coatings at your facility.

(ii) For plants without the above licensing-basis requirements, information shall be provided to demonstrate compliance with the requirements of 10 CFR 50.46b(5), "Long-term cooling" and the functional capability of the safety-related containment spray system (CSS) as set forth in your licensing basis. If a licensee can demonstrate this compliance without quantifying the amount of unqualified coatings, this is acceptable. The following information shall be provided:

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In response to GL 98-04, you provided a letter dated November 9, 1998, for Turkey Point Plant, Units 3 and 4. This submittal provided the information requested by GL 98-04. Clarification was provided during a phone call that took place on November 15, 1999. The staff has reviewed your response and has concluded that all requested information has been provided, therefore, we consider GL 98-04 to be closed for your facility. We thank you for your prompt and complete response.

If you have any questions regarding this matter, please contact me at (301) 415-1496.

Sincerely,

Kahtan N. Jabbour

Kahtan N. Jabbour, Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-250 and 50-251

cc: See next page

Mr. T. F. Plunkett
Florida Power and Light Company

TURKEY POINT PLANT

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Subject:
Letter submitting response to Generic Letter 99-02, "Laboratory Testing of Nuclear-Grade Activated Charcoal."

Body:
PDR ADOCK 05000250 P

Docket: 05000250, Notes: N/A

Docket: 05000251, Notes: N/A

MA

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L-99-240

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

RE: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Response to Generic Letter 99-02

On June 3, 1999, the NRC issued Generic Letter (GL) 99-02, "Laboratory Testing of Nuclear- Grade Activated Charcoal," to request that all licensees determine whether their Technical Specifications (TS) reference ASTM D3803-1989 for charcoal filter laboratory testing. Licensees, whose TS do not reference ASTM D3803-1989, were requested to either amend their TS to reference the ASTM D3803-1989 or propose an alternative test protocol.

Florida Power & Light Company (FPL) intends to follow the recommendations of GL 99-02 and to adopt the ASTM D3803-1989 test protocol for both Turkey Point Units 3 & 4. FPL has submitted a Proposed License Amendment to reference in the TS the ASTM D3803-1989 test protocol in accordance with the GL 99-02 requested action 2 (FPL letter L-99-239). The purpose of this letter is to summarize in the attachment FPL's response to GL 99-02 requested actions.

The last laboratory tests of nuclear-grade activated charcoal filters for Turkey Point Units 3 & 4 were completed prior to August 2, 1999. These tests were completed in compliance with the current TS test protocol, ANSI N510-1975.

The next Turkey Point Units 3 & 4 charcoal tests are scheduled in the Spring and Fall of 2000, respectively.

Please contact us if there are any questions about this submittal.

Very truly yours,

R. J. Hovey
Vice President
Turkey Point Plant

SM

Attachment

cc: Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, Turkey Point Plant

A081

PDR ADOCK 05000250

9934004 18

Turkey Point Unit 3 & 4 Response To Generic Letter 99-02 Requested Actions

Requested Action # 1:

Within 180 days of the date of GL 99-02, submit a written response to the NRC describing your current Technical Specification (TS) requirements for the laboratory testing of charcoal samples for each ESF ventilation system including the specific test protocol, temperature, RH, charcoal bed thickness, total residence time per bed depth, and penetration at which the TS require the test to be performed. If your current TS specifically require laboratory testing of charcoal samples in accordance with the ASTM D3803-1989 protocol at 30 ^{0}C [86 ^{0}F], and you have been testing in accordance with this standard, then you only need to address this requested action (i.e., no TS amendment or additional testing is required).

Response to Action # 1:

Turkey Point TS requirements require laboratory analysis of representative carbon samples from the filtering units of the Emergency Containment Filtering, Post Accident Containment Vent, and Control Room Emergency Ventilation systems. The following information describes the current Turkey Point TS surveillance requirements for the laboratory testing of charcoal samples for each of the ventilation systems impacted:

Emergency Containment Filtering System

Charcoal testing requirements for the emergency containment filters are provided in Turkey Point TS 4.6.3b. Testing is required at least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following operational exposure of filters to effluents from painting, fire, or chemical release, or (3) after every 720 hours of system operation. The test requires:

"Verifying within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with applicable portions of Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, and performed in accordance with ANSI N-510-1975, meets the acceptance criteria of greater than 99.9% removal of elemental iodine; and that any charcoal failing to meet this criteria be replaced with charcoal that meets or exceeds the criteria of position C.6.a of Regulatory Guide 1.52, Rev. 2."

ANSI N510-1975 provides instructions for laboratory testing of nuclear-grade charcoal adsorbent and specifies that adsorbent samples be tested in accordance with Table 4 of RDT M 16-1, Gas-Phase Adsorbents for Trapping Radioactive Iodine and Iodine Components. Table 4 specifies the requisite decontamination efficiency test that must be performed on the charcoal samples. Section 4.5 of RDT M16-1 specifies the test conditions that must be met. Turkey Point charcoal samples are currently tested with elemental iodine (I_2) at 130 °C and a relative humidity of 95%. Typical laboratory test conditions are provided below:

	Temperature	Relative Humidity	Test Times
Loading	130.0 °C	95%	90.0 minutes
Elution	130.0 °C	95%	90.0 minutes

Face Velocity:	12.20 m/min (40.0 ft/min)
I_2 Challenge Concentration:	17.5 mg/m^3
Pressure:	394.5 kPa (3.91 atm)

Bed Depth:	2 inches
Residence Time:	0.25 seconds

The TS acceptance criteria of greater than 99.9% removal of elemental iodine yields a penetration acceptance criteria for elemental iodine of less than 0.1%.

Post Accident Containment Vent System

Charcoal testing requirements for the post accident containment vent (PACV) filter is provided in Turkey Point TS 4.6.6b. Testing is required at least once per 18 months or (1) after any structural maintenance of the HEPA filter or charcoal adsorber housings, or (2) following operational exposure of filters to effluents from painting, fire, or chemical release in any ventilation zone communicating with the system, or (3) after every 720 hours of system operation, or (4) after replacement of a filter. The test requires:

"Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample performed in accordance with ANSI N 510-1975, meets the methyl iodide removal criteria of greater than or equal to 90% and that any charcoal failing to meet the criteria be replaced with charcoal that meets or exceeds the criteria of Position C.6.a of Regulatory Guide 1.52, Revision 2."

The PACV charcoal is tested at a temperature of 25 °C and a relative humidity of 70%. Typical laboratory test conditions are provided below:

	Temperature	Relative Humidity	Test Times
Loading	25.0 °C	70%	120.0 minutes
Elution	25.0 °C	70%	120.0 minutes

Face Velocity:	12.20 m/min (40.0 ft/min)
CH_3I Challenge Concentration:	1.75 mg/m^3
Pressure:	101.3 kPa (1.00 atm)
Bed Depth:	2 inches
Residence Time:	0.25 seconds

The TS acceptance criteria of greater than or equal to 90% removal of methyl iodide yields a penetration acceptance criteria for methyl iodide of less than or equal to 10%.

Turkey Point performs a complete changeout of the PACV carbon filter media in addition to the requisite laboratory tests. The replacement frequency coincides with the test frequency stipulated in TS 4.6.6b. The replacement charcoal meets the criteria of Regulatory Guide 1.52, Revision 2, Position C.6.a. Specifically, Position C.6.a.(2) requires new charcoal to meet the physical property specifications of ANSI N509-1976, Table 5.1. Table 5.1 describes the specific test protocol and acceptance criteria that must be satisfied by the replacement charcoal.

Control Room Emergency Ventilation System

Charcoal testing requirements for the Control Room Emergency Ventilation System filters are provided in Turkey Point TS 4.7.5c. Testing is required at least once per 18 months or (1) after 720 hours of system operation, or (2) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (3) following operational exposure of the filters to effluents from painting, fire, or chemical release in any ventilation zone communicating with the system, or (4) after complete or partial replacement of a filter bank. The test requires:

"Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, and analyzed per ANSI N510-1975, meets the criteria for methyl iodine removal efficiency of greater than or equal to 99% or the charcoal be replaced with charcoal that meets or exceeds the criteria of position C.6.a. of Regulatory Guide 1.52 (Revision 2)."

The charcoal in the control room emergency ventilation system is tested at a temperature of 25 °C and a relative humidity of 70%. Typical laboratory test conditions are provided below:

	Temperature	Relative Humidity	Test Times
Loading	25.0 °C	70%	120.0 minutes
Elution	25.0 °C	70%	120.0 minutes

Face Velocity:	12.20 m/min (40.0 ft/min)
CH_3I Challenge Concentration:	1.75 mg/m^3
Pressure:	101.3 kPa (1.00 atm)
Bed Depth:	2 inches
Residence Time:	0.25 seconds

The TS acceptance criteria of greater than or equal to 99% removal of methyl iodide yields a penetration acceptance criteria for methyl iodide of less than or equal to 1%.

Turkey Point performs a complete changeout of the control room charcoal filter media in addition to the requisite laboratory tests. The replacement frequency coincides with the test frequency stipulated in TS 4.7.5c. The replacement charcoal meets the criteria of Regulatory Guide 1.52, Revision 2, Position C.6.a.

Requested Action # 2

If you choose to adopt the ASTM D3803-1989 protocol, submit a TS amendment request to require testing to this protocol within 180 days of the date of this generic letter. The request should contain the test temperature, RH, and penetration at which the proposed TS will require the test to be performed and the basis for these values. If the system has a face velocity greater than [110] (August 23, 1999 errata) percent of 0.203 m/s [40 ft/min], then the revised TS should specify the face velocity. Also, indicate when the next laboratory test is scheduled to be performed.

Response to Action # 2

FPL submitted TS amendment requests to adopt charcoal testing using the ASTM D3803-1989 as the test protocol for conducting laboratory tests on both new and used charcoal in the emergency containment, post accident containment vent, and control room emergency ventilation system filtering units. The request contains the test temperature, RH, and penetration at which the proposed TS will require the test to be performed, and the basis for these values.

The proposed revision to the TS references the new test standard, and the appropriate acceptance criteria for maximum allowable methyl iodide penetration that must be met to satisfy the surveillance requirement. The penetration acceptance criteria proposed for the emergency containment filters and the control room emergency ventilation filter are based on the methyl iodide removal efficiencies assumed in the plant safety analysis plus a safety factor of 2. Methyl iodide penetration acceptance criteria is not currently included in the ECF TS so the test requirement represents a new license commitment. Methyl

iodide testing, however, is included as part of the control room charcoal filter surveillance test. The proposed revision reduces the safety factor from its current value of 5 down to a value of 2 to coincide with a reduction in the inherent inaccuracies associated with laboratory test standards. The post accident containment vent (PACV) filter acceptance criteria for maximum allowable methyl iodide penetration included in the PLA is derived directly from the removal efficiency for methyl iodide that is published in the current plant TS, without a change in specification safety factor.

The next laboratory surveillance test for engineered safety feature charcoal filters at Turkey Point is scheduled in March of 2000 during the upcoming Turkey Point Unit 3 refueling outage.

Requested Action # 3

If you are proposing an alternate test protocol, address the attributes discussed below and submit a TS amendment request to require testing to this alternate protocol within 180 days of the date of this generic letter. The request should contain the test temperature, RH, and penetration at which the proposed TS will require the test to be performed and the basis for these values. If the system has a face velocity greater than [110] percent of 0.203 m/s [40 ft/min], then the revised TS should specify the face velocity. Also, indicate when the next laboratory test is scheduled to be performed.

Response Action # 3

FPL is not proposing an alternate test protocol for use at Turkey Point Units 3 & 4.

Requested Action # 4

At the next required laboratory surveillance test of a charcoal sample that is 60 or more days after the date of this generic letter, test your charcoal samples in accordance with ASTM D3803-1989 or replace all of the charcoal with new charcoal that has been tested in accordance with ASTM D3803-1989. In all cases, the results should meet the acceptance criterion that is derived from applying a safety factor as low as 2 (see the note in Enclosure 2 of the GL) to the charcoal filter efficiency assumed in your design-basis dose analysis and the charcoal samples should continue to be tested in accordance with ASTM D3803-1989, in lieu of the current TS-required laboratory testing, until the TS amendment is approved by the NRC.

Response to Action # 4

The next laboratory surveillance test for Engineered Safety Feature (ESF) charcoal filters for Turkey Point Unit 3 is scheduled in March of 2000. Assuming the proposed amendments are approved or specific enforcement discretion is granted prior to that time, FPL will conduct the charcoal surveillance tests in accordance with ASTM D3803-1989. Any replacement charcoal will also meet the 1989 ASTM standard. FPL requested the approval of these amendments by February 14, 2000, to support this schedule. The charcoal test results will meet acceptance criterion that is derived from applying at least a safety factor of 2 to the charcoal filter efficiency assumed in the design basis dose analysis.

Requested Action # 5

Addressees who choose not to do the above actions are requested to notify the NRC in writing of their decision, as soon as a decision is reached but no later than 60 days from the date of this generic letter. The 60 day written response should also discuss (1) addressee plans to pursue a proposed alternative course of action (including the basis for establishing its acceptability), (2) the schedule for submitting that proposal for NRC staff review (that proposal should be submitted to the NRC no later than 180 days from the date of this generic letter), and (3) the basis for continued operability of affected systems and components until such time that the proposed alternative course of action is approved by the NRC.

Response to Action # 5

FPL plans to implement the recommendations of the GL 99-02, therefore no 60 day response was required.

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Subject:
Generic Letter 97-01,"Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations": Review of the Responses for the Turkey Point Plant, Units 3 and 4 (TAC Nos. M98606 and M98607)

Body:

Docket: 05000250, Notes: N/A

Docket: 05000251, Notes: N/A

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

January 27, 2000

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

SUBJECT: GENERIC LETTER 97-01, "DEGRADATION OF CONTROL ROD DRIVE MECHANISM NOZZLE AND OTHER VESSEL CLOSURE HEAD PENETRATIONS": REVIEW OF THE RESPONSES FOR THE TURKEY POINT PLANT, UNITS 3 AND 4 (TAC NOS. M98606 AND M98607)

Dear Mr. Plunkett:

On April 1, 1997, the U.S. Nuclear Regulatory Commission staff issued Generic Letter (GL) 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," to the industry, requesting that addressees provide a description of the plans to inspect the vessel head penetrations (VHPs) at their respective pressurized water reactor (PWR) designed plants. In the discussion section of the GL, the staff indicated that it did not object to individual PWR licensees basing their inspection activities on an integrated, industry-wide inspection program.

By letters dated April 25 and July 28, 1997, you provided your 30-day and 120-day responses, respectively, to GL 97-01. Your letter dated February 18, 1999, provided your response to the staff's request for additional information (RAI) dated October 26, 1998, relative to the issuance of the GL. The responses provided your proposed program and efforts to address the potential for primary water stress corrosion cracking (PWSCC) to occur in the control rod drive mechanism (CRDM) nozzles at the Turkey Point Plant, Units 3 and 4.

The Westinghouse Owners Group (WOG), in coordination with the efforts of the Nuclear Energy Institute (NEI) and the other PWR Owners Groups (the Babcock and Wilcox Owners Group [B&WOG] and Combustion Engineering Owners Group [CEOG]), determined that it was appropriate for its members to develop a cooperative integrated inspection program in response to GL 97-01. Therefore, on July 25, 1997, the WOG submitted two Topical Reports, WCAP-14901, Revision 0, and WCAP-14902, Revision 0, on behalf of the member utilities in the WOG. In these reports, the WOG provided descriptions of the two models, the Electric Power Research Institute (EPRI)/Dominion Engineering CIRSE Model (crack initiation and growth susceptibility model) and the Westinghouse Model, that were being used to rank the VHPs at the participating plants in the WOG. In your April 25 and July 28, 1997, responses, you indicated that you were a participant in the WOG's integrated program for evaluating the potential for PWSCC to occur in the VHPs of Westinghouse-designed PWRs, and that you were endorsing the probabilistic susceptibility model in WCAP-14902, as being applicable to the assessment of VHPs at the Turkey Point Plant, Units 3 and 4.

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The staff performed a review of your responses dated April 25 and July 28, 1997 and the applicable WCAP for your facility and determined that some additional information was needed to complete its review. Therefore, on October 26, 1998, the staff issued an RAI requesting: (1) a description of the probabilistic susceptibility ranking for a plant's VHPs to undergo PWSCC relative to the rankings for the rest of the industry; (2) a description of how the respective susceptibility models were benchmarked; (3) a description of how the variability in the product forms, material specifications, and heat treatments used to fabricate a plant's VHPs were addressed in the susceptibility models; and (4) a description of how the models would be refined in the future to include plant-specific inspection results. As was the case for the earlier responses to the GL, the staff encouraged a coordinated, generic response to the requests in the RAI.

On December 11, 1998, NEI submitted a generic, integrated response to the RAIs on GL 97-01 on behalf of the PWR-industry and the utility members in the owners groups. In the generic submittal, NEI also provided sufficient information to answer the RAIs, and emphasized that the integrated program is an ongoing program that will be implemented in conjunction with EPRI, the PWR owners groups, the participating utilities, and the Material Reliability Project's Subcommittee on Alloy 600. By letter dated March 21, 1999, the staff informed NEI that the integrated program was an acceptable approach for addressing the potential for PWSCC to occur in the VHPs of PWR-designed nuclear plants, and that licensees responding to the GL could refer to the integrated program as a basis for assessing the postulated occurrence of PWSCC in PWR-design VHPs.

To date, all utilities have implemented VT-2 type visual examinations of their VHPs in compliance with the American Society of Mechanical Engineers requirements specified in Table IWB-2500 for Category B-P components. Most utilities, if not all, have also performed visual examinations as part of plant-specific boric acid wastage surveillance programs. In addition, the following plants have completed voluntary, comprehensive augmented volumetric inspections (eddy current examinations or ultrasonic testing examinations) of their CRDM nozzles:

- 1994 - Point Beach Unit 1 (Westinghouse design)
- 1994 - Oconee Unit 2 (B&W design)
- 1994 - D.C. Cook Unit 2 (Westinghouse design)
- 1996 - North Anna Unit 1 (Westinghouse design)
- 1998 - Millstone Unit 2 (a CE design)
- 1999 - Ginna (a Westinghouse design)

In addition, the following plants have completed voluntary, limited augmented volumetric inspections of their VHPs as well:

- 1995 - Palisades - eight instrument nozzles (CE design)
- 1996 - Oconee Unit 2 - reinspection of two CRDM nozzles (B&W design)
- 1997 - Calvert Cliffs Unit 2 - vessel head vent pipe (CE design)

The majority of these plants have been ranked as having the more susceptible VHPs in the industry. Of these inspections, only the inspections at D.C. Cook Unit 2 have resulted in the identification of any domestic PWSCC type flaw indications. The current program includes

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additional commitments to perform further volumetric inspections of the CRDM nozzles at Oconee Unit 2 (a reinspection of 2-12 nozzles in 1999), Diablo Canyon Unit 2 (in 1999, a Westinghouse design), Crystal River 3 (in 2001, a B&W design), Farley Unit 2 (in 2001, a Westinghouse design), and San Onofre Unit 3 (in 2002-2008, a CE design). These plants are currently ranked in either the high or moderate susceptibility categories.

In your RAI response dated February 18, 1999, you endorsed the NEI submittal of December 11, 1998, and indicated that you were a participant in the NEI/WOG integrated program. Since the additional voluntary volumetric inspections performed to date have confirmed that PWSCC is not an immediate safety concern with respect to the structural integrity of VHPs in domestic PWRs, and since we have approved the integrated program for implementation, we conclude that the integrated program provides an acceptable basis for evaluating your VHPs. You may refer to the integrated program when submitting VHP-related licensing action submittals for the remainder of the current 40-year licensing period. However, if you are considering applying for license renewal of your facility, your application will need to address the following items: (1) an assessment of the susceptibility of your VHPs to develop PWSCC during the extended license terms for the facility; (2) a confirmation that the VHPs at your facility are included under the scope of your boric acid corrosion inspection program, and (3) a summary of the results of any inspections that have been completed on your VHPs prior to the license renewal application, as appropriate.

This completes the staff's efforts relative to your responses to GL 97-01. Thank you for your consideration and efforts in addressing this issue.

If you have any questions regarding this matter, please contact me at (301) 415-1496.

Sincerely,
/RA/
Kahtan N. Jabbour, Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-250 and 50-251

cc: See next page

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Docket Nos. 50-250 and 50-251

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