NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

DOCKET NUMBER: 50-250/251

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**DESCRIPTION**

LTR RE OUR 6-10-76 LTR.....FURN ADDL INFO TO REACTOR PRESSURE VESSEL SUPPORTS

PLANT NAME: Turkey Points 3 & 4

DISTRIBUTION FOR REACTOR VESSEL SUPPORT INFO FOR OPERATING REACTORS PER MR. TRAMMELL 7-12-76

**ENCLOSURE**

ACKNOWLEDGED

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FLORIDA POWER & LIGHT COMPANY

July 13, 1976
L-76-257

RECEIVED JUL 16 1976 U.S. NUCLEAR REGULATORY COMMISSION Mail Section

Office of Nuclear Reactor Regulation
Attention: Mr. George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 10555

Dear Mr. Lear:

Re: TURKEY POINT UNITS 3 and 4
DOCKET NOS. 50-250 and 50-251
REACTOR PRESSURE VESSEL SUPPORTS

Your letter of June 10, 1976 requested that Florida Power and Light Company (FPL) inform you of our schedule for providing an evaluation of the adequacy of the reactor pressure vessel supports at Turkey Point Units 3 and 4. Our response was to be based on a Request for Additional Information which was attached to the letter. Your request was a followup to your letter of October 17, 1975 which informed us that you were initiating a generic evaluation of reactor pressure vessel supports.

Since October 1975, FPL and several other utilities with operating Westinghouse PWRs have met to discuss how to address the adequacy of reactor pressure vessel supports in operating plants. This "owners group" considered performing analyses similar to those suggested in your Request for Additional Information either for each individual design or for several typical designs which would envelope most of the plants in the owners group. Discussions with Westinghouse Electric Corporation have indicated that such analyses would require a significant effort and, in fact, would take more than one year but probably less than three years to complete.

Plants in the design stage can include those design features which reduce the amount of analysis needed to demonstrate acceptable consequences from postulated events. Operating plants, on the other hand, cannot easily change structural configurations. Thus, events which were previously analyzed in accordance with approved techniques and found acceptable will now, because of recently developed analytical methods, require costly state-of-the-art techniques to demonstrate that the consequences remain within acceptable limits. We have therefore concluded that our most reasonable course of action in response to your questions is to propose

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implementation of an augmented inservice inspection program to preclude a reactor coolant pipe break near the reactor vessel nozzles. This program will have a positive impact on plant safety and eliminate the need to perform extensive and lengthy analyses which would have minimal impact on the real margin of safety existing in our plants.

Representatives of the owners group and the Westinghouse Electric Corporation met with menbers of the Nuclear Regulatory Commission Staff on May,25, 1976 to discuss the efforts the group has made up to that time. The discussions included justification for an augmented inservice inspection program and the technical merits of such a program. A report detailing the discussions held at that meeting is now being prepared and should be completed by September 1, 1976. Upon completion, it will be formally transmitted to you as technical justification for our selection of the augmented inservice inspection program.

Very truly yours,

Robert E. Uhrig

Robert E. Uhrig
Vice President

REU/bfp

CC: Mr. Norman C. Mosely
Jack R. Newman, Esquire