AUG 3 1977

Docket No. 50-251

Florida Power & Light Company
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Gentlemen:

Enclosed is a signed original of an Order for Modification of License, dated August 3, 1977, issued by the Commission for Turkey Point Unit No. 4. This Order amends paragraph 3.D of the Turkey Point Unit No. 4 Facility Operating License DPR-41 and permits continued operation of Turkey Point Unit No. 4 for six equivalent months of Cycle 4 operation. This Order also contains other limitations for operation of Unit No. 4.

A copy of the Order is being filed with the Office of the Federal Register for publication.

Sincerely,

Original signed by
George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosure:
Order for Modification of License

cc w/enclosure:
See next page

OFFICE	ORB#3	ORB#3	OELD	AD/ORs	DD/ORP	DD/NRR
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DATE	8/ 3 /77	8/ 3 /77	8/ /77	8/ 3 /77	8/ 3 /77	8/ /77

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Florida International University
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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
FLORIDA POWER AND LIGHT COMPANY	)	Docket No. 50-251
	)	
Turkey Point Plant Unit No. 4	)	

ORDER FOR MODIFICATION OF LICENSE

I.

The Florida Power and Light Company (the Licensee), is the holder of Facility Operating License No. DPR-41 which authorizes the operation of the nuclear power reactor known as Turkey Point Unit No. 4 (the facility) at steady state reactor power levels not in excess of 2200 thermal megawatts (rated power). The facility is a pressurized water reactor (PWR) located at the Licensee's site in Dade County, Florida.

II.

On February 8, 1977, the Nuclear Regulatory Commission ordered Turkey Point Unit No. 4 to perform an inspection of steam generators at the end of the then current fuel cycle or within 120 equivalent days of power operation from February 8, 1977, whichever occurred first. On May 3, 1977, the Nuclear Regulatory Commission issued a supplementary Order granting approval for resumption of reactor operation until the end of the third fuel

cycle and continuing the other requirements of the Order of February 8, 1977 in force. On May 9, 1977, Turkey Point Unit No. 4 was shutdown for refueling for Cycle 4 and for inspection of the three steam generators in accordance with the above Orders. The Orders required Nuclear Regulatory Commission approval before resuming reactor power operation after the shutdown.

On June 9, 1977, the Licensee submitted a report describing the results of their inspections and tests of the steam generators, as well as their analysis and evaluation of the data. The report was supplemented by letters dated June 10, 1977, June 28, 1977, July 6, 1977, July 27, 1977 and July 29, 1977. The licensee also submitted a revised ECCS performance analysis taking into account additional tube plugging. The NRC staff has evaluated this information and has assessed continued operation of the facility. This evaluation is set forth in the accompanying Safety Evaluation Report. Based on its review, the staff has concluded that operation of the facility, under the conditions previously imposed, may be continued for an additional period of six months and that the limitations contained in this Order will provide reasonable assurance that the public health and safety will not be endangered by continued operation of Unit No. 4. This Order continues in effect the leakage and radioiodine concentration limits previously imposed and makes slight revision to the peaking factor limits presently contained in the facility Technical Specifications to reflect the revised ECCS analysis.

Copies of the following documents are available for public inspection in the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. 20555 and at the Environmental and Urban Affairs Library, Florida International University, Miami, Florida: (1) the licensee's report of the steam generator inspections dated June 9, 1977 as amended on June 10, 1977 and June 28, 1977, (2) the Order for Modification of License, In the Matter of Florida Power and Light Company (Turkey Point Plant Unit No. 4), Docket No. 50-251 dated February 8, 1977, (3) our Safety Evaluation Report dated February 11, 1977, applicable to our Order dated February 8, 1977, (4) the Order for Modification of License, In the Matter of Florida Power and Light Company (Turkey Point Plant, Unit No. 4), Docket No. 50-251, dated May 3, 1977, (5) the licensee's letter of July 6, 1977 requesting approval to resume power operation of Turkey Point Unit No. 4, (6) the licensee's letters of July 27, 1977 and July 29, 1977 which evaluate operation with one or more steam generator tube plugs in the primary coolant system, and (7) this Order for Modification of License, In the Matter of Florida Power and Light Company (Turkey Point Plant, Unit No. 4), Docket No. 50-251.

III.

Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, and the Commission's Rules and Regulations in 10 CFR Part 2 and 50, IT IS

ORDERED THAT paragraph 3.D of Facility Operating License No. DPR-41 is hereby amended by the following new provisions:

D. Steam Generator Operation

1. Turkey Point Unit 4 shall be brought to the cold shutdown condition in order to perform an inspection of the steam generators after six equivalent months of Cycle 4 operation. Nuclear Regulatory Commission approval shall be obtained before resuming power operation following this inspection. For the purpose of this requirement, equivalent operation is defined as operation with a primary coolant temperature greater than 350° F.

2. Primary to secondary leakage through the steam generator tubes shall be limited to 0.3 gpm per steam generator. With any steam generator tube leakage greater than this limit, the reactor shall be brought to the cold shutdown condition within 24 hours. The leaking tube(s) shall be evaluated and plugged prior to resuming power operation.

3. The concentration of radioiodine in the primary coolant shall be limited to 1 microcurie/gram during normal operation and to 30 microcuries/gram during power transients.

4. Reactor operation shall be terminated and Nuclear Regulatory Commission approval shall be obtained prior to resuming operation if the reactor is required to shutdown due to primary to secondary leakage, as specified in paragraph 2, above, more than twice during a 20 day period.

5. The operation of the Metal Impact Monitoring System (MIMS) with the capability of detecting loose objects will be continued until the next reactor vessel inspection. In the event that the MIMS is out of service, it will be reported to the NRC. Any abnormal indications from the MIMS will also be reported to the NRC by telephone by the next working day and by a written evaluation within two weeks.

6. Following each startup from below 350°F, core barrel movement will be evaluated using neutron noise techniques.

7. On page 3.2-3 of the Technical Specifications for Turkey Point Unit No. 4, the peaking factor, Fq, is hereby reduced from 2.22 to 2.20 at rated power. This

change will be incorporated in the Technical Specifications in a future amendment responding to your submittal of June 8, 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation

Dated in Bethesda, Maryland
this 3rd day of August 1977.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING ORDER FOR MODIFICATION OF LICENSE

FLORIDA POWER AND LIGHT COMPANY

TURKEY POINT NUCLEAR GENERATING UNIT NO. 4

DOCKET NO. 50-251

Introduction

On February 8, 1977, we issued an Order for Modification of License for Turkey Point Unit No. 4. Our Order amended Facility Operating License DPR-41 and permitted continued operation of Unit No. 4 for 120 equivalent days from February 8, 1977, or until the end of the then current fuel cycle, whichever occurred first, at which time Unit No. 4 was to be brought to a cold shutdown condition for inspection of the three steam generators. The Order contained implementing requirements and limitations on operation. The Order also required Nuclear Regulatory approval before resuming reactor power operation after the shutdown for inspection of the steam generators. The Order of February 8, 1977 was supported by our Safety Evaluation dated February 11, 1977. On May 3, 1977, we issued another Order for Modification of License for Unit No. 4 continuing the Order of February 8, 1977 in force until the end of fuel cycle 3.

On May 9, 1977, Turkey Point Unit No. 4 was shutdown for refueling for Cycle 4 and to inspect the steam generators. In accordance with our Orders, Florida Power and Light Company (FPL) performed an extensive inspection of the three steam generators. The results of the inspection were presented in a report forwarded by the licensee's letter of June 9, 1977. The report was supplemented by letters dated June 10, 1977 and June 28, 1977. The report presented a comprehensive analysis and evaluation of the rate of tube degradation during the next year of operation. Criteria were developed for plugging tubes, under which a total of 590 additional tubes were plugged during the outage. With the implementation of this plugging criteria, the licensee concluded that Unit No. 4 was expected to operate safely and with minimal leakage for a period in excess of six months. Accordingly, in their letter of July 6, 1977, FPL requested an amendment of the Turkey Point Unit No. 4 Facility Operating License to permit Unit No. 4 to return to power.

Discussion

As a result of the additional number of tubes that were plugged in the three steam generators, the licensee submitted a reevaluation of the Emergency Core Cooling System (ECCS) performance for Turkey Point Units Nos. 3 and 4. This reevaluation was performed for an average of 15 percent of the steam generator tubes plugged, using the assumptions employed in the previous ECCS reevaluation which has been reviewed and approved by the staff. A request was also made by the licensee to decrease the value of the peaking factor, F_q, in the Technical Specifications from 2.22 to 2.20 whenever steam generator tube plugging exceeded 10 percent. The reason for this reevaluation is to allow the plants to be operated with more than 10 percent of the steam generator tubes plugged. In addition to the ECCS evaluation, the staff evaluated the inspections and analysis performed by the licensee on the steam generators.

On May 24, 1977, we were notified that there was a missing plug in the cold leg of a tube in the Unit No. 4 B steam generator. Subsequently, we were notified that there were a total of 11 such missing plugs in the B steam generator, all in Row 1. All of the Row 1 tubes were supposedly plugged for preventative purposes during the September and November 1976 shutdowns.

Appendix A discusses the staff's evaluation of the ECCS performance with up to an average of 15% of the steam generator tubes plugged. Appendix B discusses the staff's evaluation of the steam generator inspection and plugging operation. Appendix C discusses the potential safety implications of operation with loose plugs in the reactor coolant system.

Environmental Considerations

We have determined that the revised operating limitations do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this action.

Conclusions

We have concluded, based on the considerations discussed above, that with respect to the return of Unit 4 to full power operation there is reasonable assurance that the health and safety of the public will not be endangered by operation of Unit 4 in the proposed manner. We have also concluded that operation with a foreign object in the reactor vessel will not result in any adverse structural effects on the reactor vessel, its cladding or reactor internals and that the presence of this object will not create any adverse thermal-hydraulic or other core conditions. Furthermore, we have concluded that Unit 4 will be operated in compliance with the Commission's regulations and that issuance of this Order will not be inimical to the common defense and security or to the health and safety of the public.

APPENDIX A

SAFETY EVALUATION REPORT

TURKEY POINT NO. 4

ECCS PERFORMANCE EVALUATION

ECCS Performance Evaluation

The ECCS analysis provided by Florida Power and Light Company for Turkey Point Units Nos. 3 and 4 consisted of an evaluation of ECCS performance using the October 1975 version of the Westinghouse ECCS evaluation model and included the following assumptions:

(a) 15 percent steam generator tubes plugged

(b) Maximum Peaking Factor, $F_q = 2.20$

(c) Accumulator minimum water volume: 875 ft^3

(d) Upper head fluid temperature equal to reactor vessel outlet (hot leg) temperature

(e) Coolant inlet temperature, $T_{in} = 550.2^oF$

The analysis was performed for a double ended cold leg guillotine break (DECLG) with discharge coefficient, $C_d = 0.4$. This break size was identified as the critical break in the previous evaluation (References 1 and 2) performed using the same assumptions except for steam generator tube plugging which was assumed to be 10 percent. The licensee has demonstrated (Reference 3) that the critical break size will remain unchanged for 15 percent of the steam generator tubes plugged.

The results of the analysis indicate that the limiting values for peak clad temperature and local Zr - water reaction are: 2173^oF and

11.655 percent, respectively. Both these values are below the limits specified in 10 CFR 50.46.

The sensitivity study performed in the previous ECCS analyses considered the effect of the degree of steam generator tube plugging on peak clad temperature and local Zr - water reaction. It was demonstrated that both these parameters increase with increasing percentage of plugged tubes. The applicability of the present analysis is limited, therefore, to operation of Turkey Point Units Nos. 3 and 4 with up to 15 percent plugged tubes. Similar restriction would apply to the maximum peaking factor (F_q). For 10 percent or less of the steam generator tubes plugged F_q = 2.22 and for plugging values between 10 and 15 percent F_q = 2.20. Whenever the plugging exceeds 15 percent a new ECCS analysis would have to be performed.

Recently the staff has noted that in LOCA calculations for some PWRs, a decrease in primary coolant inlet temperature has resulted in a predicted increase in peak clad temperature. In discussions with the PWR vendors we have learned that they have all observed this trend while performing LOCA calculations with their individual approved evaluation models. In the past, it has been widely accepted that it was conservative to assume the highest possible initial coolant temperature for LOCA calculations (typically maximum full power operating temperature plus 4°F for measurement uncertainty).

The apparent cause of this behavior stems from the fact that a reduction in coolant inlet temperature results in a reduction in the coolant saturation pressure. This decreases the flow rate from the vessel side of the break after the short period of subcooled blowdown. This reduced flow, for the postulated cold leg break, decreases the magnitude of the downward flow rate through the core that exists for a large portion of the blowdown period. This decreases the heat transfer coefficient and consequently less stored energy is removed during blowdown.

Reducing the coolant inlet temperature also changes the flow rate from the top of the vessel to the hot leg and out of the break through the steam generator and reactor coolant pump. The changes in hot leg flow caused by a reduction in inlet temperature tend to decrease the core flow rate during the period of positive core flow. This also leads to the removal of less stored energy during blowdown. Thus, the fuel temperature is higher at the end of bypass. Most PWRs exhibit peak clad temperature during reflood, and entering the reflood period with a greater fraction of stored heat remaining after blowdown may cause an increase in the peak clad temperature. It has also been observed that the decreased negative core flow may extend the time to end of bypass. Then in the evaluation model more accumulator water is assumed to spill out of the break. If, as a result, there is insufficient accumulator water remaining to fill the downcomer, reflood will be delayed. This will also contribute to the increase in peak clad temperature.

However, a reduction in coolant inlet temperature may not always result in an increase in peak clad temperature. It has been observed that if the clad rupture location changes to a different elevation where the core power is less, peak clad temperature may decrease.

In addition to the predicted changes in blowdown core flow and heat transfer, reducing coolant inlet temperature also causes a slight reduction in containment back pressure during reflood. Reducing this pressure is known to result in lower reflood rates with correspondingly higher clad temperatures. However, the effect due to containment back pressure is minor compared to blowdown core flow and heat transfer effects.

At this time the staff believes that nominal values of inlet temperatures and steam generator shell side steam conditions should be used in all LOCA calculations since the effects of variations in inlet temperatures and steam conditions on peak clad temperature are not consistent and are at best second order effects.

The maximum sensitivity of peak clad temperature to inlet temperature that the staff has seen to date shows that for a $1^{o}F$ decrease in inlet temperature the peak clad temperature following a large break LOCA would increase $4^{o}F$. However, a reduction in inlet temperature results in a corresponding reduction in core average temperature and steam generator shell side steam pressure. A reduction in steam generator secondary pressure results in lower peak clad temperatures. If we now assume a decrease in coolant inlet temperature and adjust the steam generator shell side steam conditions, the peak clad temperature for Turkey Point would increase less than $20^{o}F$. Thus we consider this a minor second order effect. We conclude that the current ECCS analysis on file for Turkey Point meets the criteria of 10 CFR 50.46 and is therefore acceptable. However, our review of the matter has enabled us to identify areas in which there may be additional improvements in accuracy of modeling coolant inlet temperature and steam generator steam conditions. We are currently seeking additional information

to be used in a generic evaluation of the effect of coolant inlet temperature and steam conditions on ECCS performance.

The licensee has agreed to submit an ECCS analysis of the worst break LOCA assuming coolant inlet temperature and steam conditions equal to their nominal values.

Nominal inlet temperature and steam conditions as used here refer to the most probable values for the plant when it is operating at 102% power.

This will confirm the applicability of the peak clad temperature sensitivity, stated above, to the Turkey Point Plants and the effect of the use of most probable values for these nominal conditions.

Conclusion

Based on our review, we conclude that the Turkey Point Plants' ECCS performance will conform to the peak clad temperature, maximum local oxidation, hydrogen generation, coolable geometry and long term cooling criteria of 10 CFR 50.46 provided that the peaking factor, F_q, does not exceed 2.22 for up to 10 percent of the steam generator tubes plugged and 2.20 for between 10 and 15 percent of the tubes plugged. For operation of the plants with more than 15 percent tube plugging a new analysis would have to be provided.

References

1. Letter from Robert E. Uhrig, FPL, to NRC, L-76-419, dated December 9, 1976, transmitting "Major Reactor Coolant System Pipe Rupture Analysis".

2. Letter from Robert E. Uhrig, FPL, to NRC, L-77-1, dated January 3, 1977.

3. Letter from Robert E. Uhrig, FPL, to NRC, L-77-217, dated July 11, 1977.

APPENDIX B

SAFETY EVALUATION REPORT

TURKEY POINT NO. 4 STEAM GENERATORS

DOCKET NO. 50-251

INTRODUCTION

By Order dated February 8, 1977, the Nuclear Regulatory Commission ordered Turkey Point Unit No. 4 to perform an inspection of steam generators at the end of the last fuel cycle or within 120 equivalent days of power operation from February 8, 1977, whichever occurred first. Among other operational limitations, the NRC Order specifically required that during this period the reactor operation shall be terminated if primary to secondary leakage which is attributable to two (2) or more tubes per plant occurs during a twenty (20) day period. Nuclear Regulatory Commission approval was required before resuming reactor power operation after such a shutdown.

On March 20, 1977, the unit was shutdown to plug a leaking tube in steam generator C of Unit No. 4. During this outage, a second leaking tube was discovered and was also plugged. The tube leaking incident was first observed in mid-February and progressed very slowly over a period in excess of one month. The leakage behaved in a predictable fashion and had no safety consequences that were not previously evaluated.

After discussions with the NRC staff, with respect to the licensee's assessment that continued facility operation with the identified leaks plugged would not endanger public health and safety and did not require specific approval under the provisions of the Order, the Unit was returned to operation on March 25, 1977.

On April 25, 1977, the licensee informed the NRC that they had detected another leak with an equivalent leakage rate of about 0.04 GPM. By April 27, 1977, the leakage rate had progressed to 0.14 GPM and the Unit was shutdown for investigation. On April 28, 1977, the NRC staff was informed that three leaking tubes were identified in the C steam generator (row 2 - column 47, row 2 - column 61 and row 3 - column 62). The

information developed by the licensee's inspection indicated that the leaks were attributed to tube denting. The leaking tubes were located in "hardspot" regions between flow slots where tube denting was predicted to be more severe than in other areas of the tube bundle. The elevations of these leaks were determined to be at the fourth and the fifth support plates. By letter dated April 29, 1977, the licensee submitted: (1) results of their inspection and safety evaluation of the three leaking tubes, and (2) their corrective actions and bases to resume power operation. In addition, the licensee requested NRC approval to continue power operation for the remaining fuel cycle, which was estimated to be about fifteen (15) equivalent days. The NRC staff reviewed the submitted information and concurred that the resumption of power operation by Turkey Point Unit No. 4 would not present a significant risk to the public health and safety.

All leaks associated with dented tubes experienced up to this point in time had been small, well below the leakage limits established by license condition or Technical Specifications. The leakage rate progressed slowly and was detectable. Tube cracks which result from severe denting were constrained within the tube support plates; and, thus, any leaks caused by this type of crack would be limited even under accident conditions. Therefore, on May 3, 1977 continued operation for the remaining fuel cycle was approved for Unit No. 4. Approximately one week after returning to power operation, one more tube leak was discovered. FPL then made the decision to shutdown Unit No. 4 to perform the scheduled refueling operation and conduct the scheduled inspection of the tubes in all three steam generators.

DISCUSSION

Inspection Program

Of more significance in the long term was the need to carefully assess the condition of all three steam generators in Unit No. 4 and to determine, to the extent possible, the causes for the continuing occurrences of leakage in the facility. By letter dated April 11, 1977, FP&L proposed to conduct a thorough inspection and evaluation of the steam generators during the refueling outage. The proposed program was reviewed by and discussed with the NRC staff to assure staff concurrence with the program. In this connection, FP&L originally attributed the March leak to corrosive conditions in the area between the first tube support plate and the tubesheet, whereas the most recent leaks in May were attributed to denting. Since these two different conditions required different assessments and treatment, it was important to identify the cuases of leaks which had occurred. After discussions with the staff, the licensee committed to pull and metallurgically evaluate at least one tube from Unit No. 4 steam generators. The entire tube was to be pulled and metallurgically examined at each area of suspected degradation, i.e., at each tube/tube sheet or tube/support plate intersection. The tube selected was R45C53 in Steam Generator C. If this tube could not be removed, a tube which had experienced a leaking dent was to be pulled and metallurgically examined, as described above.

Details of the steam generator inspection program as actually conducted are summarized as follows:

1. Standard eddy current techniques were used to detect sludge

accumulation on the tube sheet to determine possible areas where wastage could occur. A representative sampling of tubes in each generator from rows 7, 14, 21, 28, 33 and rows 38 through 45 between columns 15 and 78 were tested in this manner.

2. A complete Regulatory Guide 1.83 inspection program was performed in each of the three steam generators.

3. Eddy current inspections were made in all three generators in the one O'clock and eleven O'clock wedge regions to detect flaws in the tubes. These inspections were made using a standard probe, and selective use of the newly developed helical scan probe was made in generators B and C.

4. An eddy current dent location program was performed using a standard 0.540 inch probe at reduced gain. A comprehensive sampling was conducted for steam generator C, and a representative sampling was performed in each of the remaining two steam generators.

5. Tube gauging was conducted on all three steam generators utilizing three different size eddy current probes; i.e., 0.540, 0.610, and 0.650 inch probe diameters. Tube R2C49 in generator C was removed for laboratory examinations. This tube was pulled because it contained at least one intersection that did not allow passage of the 0.540 inch probe, and was selected in lieu of the tube that leaked previously.

6. A determination of the location and the cause of the leak in steam generator C tube R45C53 was made by extensive eddy current probing of tubes adjacent to the leaking tube, between the tube sheet and the first tube support plate. Tube R42C53 in generator C was removed for laboratory examinations in place of R45C53, because of limited access (R45C53 is located at periphery of the tube bundle). Tube gauging was also performed in this area, using the three different size probes, for denting. When Eddy current testing revealed flaws located just above the tubesheet in several tubes

near the edge of the tube bundle periphery, the straight length of an affected, accessible tube was pulled; i.e., R42C53.

7. Handhole inspections using TV and/or photographs were made to inspect the flow slots and determine the extent of hourglassing on the #1 tube support plates in steam generators A, B and C.

8. TV and/or photographs were used to inspect the flow slots and determine the extent of hourglassing on the #6 tube support plate in steam generator B.

9. Wrapper to shell annulus measurements were made on steam generators B and C.

10. Condition of tube support plates in steam generators A, B and C were assessed by Eddy current mapping using 3-1/2 KHz frequency. More than 300 tubes per S/G were probed.

11. Eddy current testing of U-bends were made in steam generator B; Rows 2-5, columns 1-92 - were tested at 100 KHz using beaded flex 0.540 probe. Any tube(s) which gave an indication at 100 KHz were reprobed using 400 KHz.

Results of Inspections and Corrective Actions

By letters dated June 9, and 28, July 10 and 15, 1977, FPL submitted results of the steam generator inspections and the laboratory examinations of two tube samples removed from S/G-C, and corrective actions implemented following the inspections. The following paragraphs summarize the results of the inspections:

The sludge heights in areas of all three steam generators were consistent with previous measurements made in 1976. The area between row 38 through 45, columns 15 and 78 is not routinely probed, but was probed during this inspection.

The sludge heights are in the order of 1/2 to 1.0 inch, which is not considered to be a problem. Although the results of the laboratory

examination of tube at R42C53 location in S/G-4C indicated that residual phosphates in the sludge right above the tube sheet caused continued tube wall thinning, but at a reduced rate of corrosion attack. This has also been confirmed by the RG 1.83 inspection, which detected 12 defective tubes that required plugging.

All previously unplugged tubes in rows 2-5 in S/G-4B were inspected through the U-bends. No discernable flaw indications were found. Results of the flow slot examinations showed that S/G-4B had the worst flow slot deformation (hourglassing) among the three steam generators. In fact, the six flow slots that could be observed in the second and the fifth support plates in S/G-4B are now completely closed. Cracking of the flow slots edges has occurred in some slots in all three generators.

Shell wrapper annulus measurements were made in S/G-4B and -4C. No detrimental demensional changes were noted. The data collected provides representative information to be used in the future for correlating predicted and actual support plate expansions. the dented tube sample removed from S/G-4C contained five support plate intersections. The non-destructive phase of the examination consists of photo documentation, eddy current examination, radiography and dimensional characterization. Eddy current examination showed apparent cracking at the fourth support level. Eddy current signals at all support regions showed slight material density variations, an indication of the formation of magnetite.

Tube R42C53 with wastage indications in the "C" generator was removed to some distance above the tube sheet, but not including the first support. Eddy current examination showed a low volume, OD initiated indications of about 35% part-through wall in an area just above the tube sheet. OD dimensional traces also showed an average loss of wall thickness of approximately 5 mils or about 10% of the wall thickness at the top of the tube sheet. Tube wall thickness measurements taken in a region that appeared to have thinned down showed a loss of wall thickness of about 13 mils in a localized area. Metallographic examination in the area near the localized thinned down region showed wall thickness loss of about 5 mils. This area was covered with black corrosion products, suggesting that active

corrosion had taken place. This corrosion attack had apparently occurred during the period when phosphate chemistry was maintained and probably continued at a reduce rate for some period after the conversion to AVT due to the residual phosphates in the sludge.

The ID gauging for denting was performed utilizing a series of different probe sizes; i.e., 0.540, 0.610, and 0.650 inch in probe diameter. The areas probed were chosen on the bases of the analysis of the critical strain contours in the tube support plate and the predicted growth of magnetite at tube/tube support plate annulus, taking into account the effect of support plate cracking. The tubes were initially gauged with the 0.650 probe. Those tubes that did not allow passage of this probe were then gauged with the 0.610 inch probe. Any tubes which did not allow passage of the 0.610 inch probe were then gauged with the 0.540 inch probe. The results of this gauging process did indicate a fair correlation with the strain analysis in the support plate. Severely dented tubes were plugged in accordance with the tube plugging criteria stated below.

Plugging Criteria

The licensee has implemented the following plugging criteria to justify a period of six (6) months operation:

1. All tubes that did not allow passage of the 0.540 inch probe were plugged.
2. To justify the six (6) month operation the following tubes were also plugged;
   a. two (2) tubes beyond (i.e., higher row number) any tube, in column 14 to 80, which did not pass the 0.540 inch probe were plugged,
   b. a maximum of four (4) tubes and a minimum of two (2) tubes beyond any tube, in columns 1 to 13 and 81 to 94 near the tube lane, that did not pass the 0.540 inch probe were plugged, and
   c. tubes at the plate perimeter near the 3 and 9 o'clock wedges (one tube on each side of a tube that did not pass the 0.540 inch probe).

3. All tubes which did not pass the 0.610 inch probe were plugged.

4. The tubes in any column for which plugging under criteria (1) or (2) or (3) above was implemented were also plugged in the lower row numbered tubes back to the tube lane if not already plugged; unless the required plugging would connect two apparently unrelated regions.

5. Additional preventive plugging was implemented for tubes R45C39 and R45C54 in each steam generators, if these tubes were not already plugged. This action was taken to alleviate concern for those unique tube locations:

   a. At the perimeter of the plate

   b. Adjacent to a discontinuity (cut) in the plate created by the installation of the patch plate

   c. Adjacent to plug welds in circulation holes.

Criteria stated in (2) for plugging the tubes beyond those which do not allow the passage of a 0.540 inch probe is based on the change in the region of tubes which are severely dented. This region is bounded by the plate strain contour that corresponds to a 12% hoop strain in the tubes. Based on the history of previous leak locations with the exclusion of the patch plate leaks, the growth of the plate strain intensity contour corresponding to the 12% to 16% hoop strain range (i.e., the range in which over 90% of the leakers have occurred) is conservatively estimated to be about one third of a tube row per month.

The implementation of above plugging criteria combined with previous plugging for various causes resulted in 10%, 17.5%, and 11.5% of tubes in steam generators A, B, and C, respectively, being plugged.

"Islanding" Effect

With respect to the possible loss of lateral support on some inner row tubes due to the so-called "islanding"' i.e., broken support plate pieces moving into flow slots the concern may be alleviated by the fact

that most inner row tubes of the tube bundle were plugged. The licensee has also submitted results of an analysis of fluid structural vibrations considering loss of one, two and three laterial supports. The maximum vibration amplitude is calculated to slightly exceed one half of the gap between adjacent tubes for the case when three lateral supports were lost, with a slight increase in bending stress in the tube.

Potential for U-Bend Cracking

As a part of the submittal on June 9, 1977, the licensee provided an additional analysis to show that indeed the residual hoop and axial stresses for tubes in rows 2 and beyond decrease as the U-bend radius increases. The magnitude of the residual stresses are dependent on the degree of cold working of the tube cross-section during the forming of the small radius U-bend. Bending of small radius U-bend tubes causes their cross-section to ovalize. However, the presence of an internal ball mandrel during bending of tubes for rows 1 and 2 forces the cross-section back into a configuration with considerable less ovality. Upon removal of the ball, the cross-section rebounds elastically and the residual hoop stresses on the inside surface of the tube are tensile at the top and bottom positions of the U-bend apex. Bending of tubes for rows greater than 2 does not employ the ball mandrel. Hence those tubes adopt the characteristic oval shape after bending, the tube cross-section rebounds elastically and the corresponding residual hoop stress for tubes in row 3 and beyond are compressive. The degree of residual tube ovality is a function of the bend radius, with the greatest ovality associated with the tubes with the smallest bend radii, i.e., row 1.

The analyses indicate that the residual hoop stresses at the U-bend apex are equal to one half of the yield strength, or about 26,000 psi and the residual axial stress in small radius U-bend tubes is 15,000 psi. The magnitudes of these residual stresses are less than the ASME Code minimum yeild strength. Therefore, with these low residual stresses at the U-bend apex, the potential for stress assisted intergranular attack at the U-bend for tubes in rows 2 and beyond is minimal. Furthermore, the fact that most inner row tubes were already plugged should alleviate concern over the potential for U-bend cracking.

Alleged Missing Tube Plugs

During a routine entry into the B generator cold leg channel head for Eddy current probing of the steam generator tubes, eleven (11) tube plugs at locations R1C1 through C9, R1C23, and R1C31 were found missing. These eleven (11) tube locations were logged as being explosively plugged in November and September, 1976. After a review of the documentation and assurances of the personnel associated with the plugging efforts in September and November of 1976, the licensee and the vendor initiated a search for plugs at possible locations in the reactor coolant system. Locations investigated include the lower portion of the reactor vessel, the spray line to the pressurizer, reactor internals, the "B" loop cold leg and the thermal shield ledge. The initial searching effort also involved removal of nine fuel assemblies from the center of the core. A miniature underwater television camera was utilized to examine the lower internals, the core support plate, and the bottom portion of the reactor vessel. No traces of plugs were found. Then, the remaining fuel assemblies were removed to allow viewing of the entire lower area of the vessel. There was no indication of any plugs or component damage. The thermal shield ledge was inspected by removing the flange plugs for access to the materials surveillance specimens. No tube plugs were detected on the ledge.

The cold leg piping of loop "B" was examined by the utilization of an underwater camera mounted on an air-operated skate board. This arrangement permitted examination of the cold leg piping between the S/G and RCP. Once again, no plugs or foreign objects were observed.

The second location that was credible for a plug to "hide-out" was postulated to be the spray line from the steam generator to the pressurizer. All potential "hide-out" locations in the pipe were radiographed, and the results indicated that there were no plugs in the pipe.

The possibilities of plug hide-out within the reactor coolant pump were also investigated. The probability that a 6 ounce, 0.775 inch diameter, 6 inch long plug will lodge in the pump is rather remote. The

smallest vane-to-vane dimension in the impeller is 4-1/2 inches, meaning that an improbable orientation of the plug is required when passing through the impeller (or defuser) to permit lodging in the vanes. This occurrence is judged to be very unlikely and, considering multiple occurances, is not deemed credible. Impeller seal ring clearances are .025 to .100 inch (assuming some wear) and are, therefore not large enough for a plug to enter. Since the diffuser exit velocity is 30 fps, it seems most likely that the plug will readily reach the discharge nozzle and be carried down the pipe. Even assuming the plug could fall to the bottom of the casing after exiting from the diffuser, the water velocity at the bottom of the casing around the casing O.D. adapter is estimated to be 15 fps which would be sufficient to pick up the plug and move it into a higher velocity area near the discharge nozzle from where it could enter the pipe.

Results of Plug Search and Remedial Actions

The results of the extensive search in the reactor coolant system revealed no loose plugs which led to the conclusion that the plugs were never installed, and/or remotely, that all or some of them might be secured somewhere in the reactor coolant system. These two possibilities are addressed by implementing the following remedial measures:

(1) Plugs Never Been Installed

a. Current plugging practices were totally reviewed by both Westinghouse and Florida Power & Light; particularly with respect to improving plug accountability and verification that plugs are properly installed in the correct tubes. Procedures would be revised to incorporate the improved methods.

b. Permanent documentation of the "as plugged conditions" would be required at the conclusion of a plugging operation. A photographic method was utilized to provide the required documentation.

(2) Plugs Are Secured in RCS

a. A metal impact monitoring system was installed on the reactor vessel. This system will detect the presence of "loose parts" in the vessel. If upon start-up, or during operations, loose parts are detected, then continued operation of the unit will be evaluated. Safety analysis to address operations with loose plugs in the RC System has concluded that the unit can be safely operated with 4 or less loose plugs.

b. Start-up surveillance requirements will be in effect so that particular attention is given to reactor vessel "noise" monitoring and reactor coolant pump behavior.

Potential Safety Implications With Loose Plugs In RCS

Loose tube plugs which enter the RCS from the outlet (cold leg) side of the steam generator can remain in the steam generator channel head or be pushed along by the reactor coolant flow through the crossover leg piping. No measurable impediment to flow is expected by the presence of one or several tube plugs in the flow path. Flow velocity in the piping is sufficient to carry plugs along to the reactor vessel.

Passage of a plug through a reactor coolant pump during operation was also considered. Westinghouse experience on canned rotor pumps in the shop has shown that if material which is relatively large (compared to the size of the impeller/diffuser vanes), breaks from the system and passes through or lodges in the pump, that the most noticeable effects will be higher vibration level (if an object lodges in a vane of the impeller) and vane denting or gouging, usually very localized. Since the plug relative size and weight is small no significant damage to the impeller or diffuser would be expected in the event a plug were to pass through the pump hydraulics.

A loose plug bounding around the bottom of the casing will have no detrimental effect on the casing interior surfaces. The casing itself is 7-1/2 inches thick (minimum) stainless steel and is unclad so there is

no problem of cladding abrasion. Furthermore, there is no delicate instrumentation of fragile hardware within the volute which has any chance of being damaged by a moving plug.

Once in the vessel, the plug will follow the path of the cold water and end up in the lower plenum. The presence of the plug in the lower plenum presents two potential problems. The first aspect considered was the potentiality of loose pieces impacting the lower internals components. Due to the low velocities present, only insignificant effects of such impact can be expected. The plugs are too large to enter either the core region or the drive line area. The chance that the plug will fracture and small pieces will migrate into the core region is believed to be remote due to the shape of the plug and material used (Inconel).

A second potential problem which may occur is wedging of the tube plugs in close clearance areas during start-up after a cold shutdown. Only one area was identified as a potential problem should wedging occur. The identified area is the clearance between the vessel and the secondary core support base plate. Should a plug become wedged between the base plate and vessel during start-up from cold shutdown, the constriction against thermal growth would cause forces to exist. The existance of such forces acting with the normal hydraulic forces could cause movement of the vessel internals. A sensitivity study indicated that for the most conservative conditions more than four plugs would need to be wedged between the vessel and the baseplate. The forces generated by any lesser number of wedged plugs will not be large enough to cause movement of the internals.

The potential effects of loose plugs on the fuel performance and integrity were also evaluated. Since it was previously mentioned that no plugs could enter the core through the lower internals, no nuclear, thermal, hydraulic, or mechanical concerns exist. Even if plugs were found on the top surface of the core, no nuclear, thermal, hydraulic, or mechanical damages are expected. When operation commences, any plugs on the top of the core would probably be swept down the hot leg and into a channel

head where it would come to rest in a low flow area. The clearances between fuel pins and assemblies are too small to allow movement of a plug into those areas.

The licensee has demonstrated that only the wedging of a plug between the secondary core support base plate and the vessel is a potential problem of operation with loose plugs in the RCS, and greater than 4 loose plugs are required to pose a safety concern. Wedging of loose plugs between the reactor vessel and the secondary core support base plate can induce sufficient force during heat-up to cause excessive vibration of the reactor internals during normal operation. However, any excessive movement of the reactor internals can be detected by the ex-core neutron noise monitor to initiate plant shutdown and corrective action and, thus, the safety consequences would be limited. NRC's determination of the safety implication is contained in the following section.

EVALUATION

By letter dated July 6, 1977, the licensee (FPL) proposed to start Cycle 4 operation of Turkey Point Unit No. 4 for a period of six (6) months. This proposal was based upon the result of the extensive examination program and the supporting conclusions discussed above. The NRC staff has reviewed the information submitted by the licensee and concurs in the following:

1. Further U-bend failures are not likely to occur for near term continued operation because of the following:

   a. Laboratory examinations of 71 tubes removed from Surry Unit Nos. 1 and 2 and Turkey Point Unit No. 4 steam generators indicate that cracking was confined only to row one tubes.

   b. All the tubes in row one and most tubes in row two are plugged.

   c. The susceptability for U-bend cracking of tubes in row 2 would be substantially less because the residual hoop stress at the U-bend apex is 25-35% lower than the ASME Code minimum yield strength for Inconel 600 alloy tubing and the effective U-bend strain is 30-50% lower than in row 1.

d. U-bend cracking in rows 3 and beyond will not occur because the residual hoop stresses on the inside surface of the tube are compressive at the top and bottom positions of the U-bend apex, and thus the potential for stress corrosion cracking is not possible.

2. Support plate expansion or continuing magnetite growth in the proposed period of operation will have insignificant effects on the wrapper and the steam generator vessel. Therefore, the wrapper and the vessel integrity during normal operating and accident conditions will not be affected by continued support plate expansion.

   Due to the closure of the flow slots in the top support plates and the possible closure of flow slots in lower support plates, additional loads could be transmitted to the steam generator shell through the load path of the support plate, wedge, wrapper and channel spacer. Based on preliminary "crush" tests performed by Westinghouse the maximum load that can be developed along this load path is 60,000 pounds.

   Analysis of the bearing stress along this path indicates that all stresses are less than the yield strength. Such stresses on the steam generator shell are highly localized and self limiting and will not adversely affect the integrity of the shell under accident conditions.

3. With respect to the effect of continued magnetite growth which causes the support plate expansion and thus denting, the preventive plugging program that was implemented in accordance with the criteria discussed previously is adequate. In this regard, the NRC staff also considered the following additional supportive reasons:

   a. Refined strain analyses of the support plate expansion (to complete flow slot closure) indicated small strain or deformation increases in hard spot regions around the flow slots and plate perimeter.

b. All of the tubes in hard spot regions and those that have leaked previously have been plugged, based on the criteria derived from past experience and the correlation between the strain predictions and field gaging results. This corresponds, in general, with the calculated strain pattern due to a conservatively estimated magnetite growth rate.

c. All leaks associated with dented tubes experienced to date have been small; i.e., well below the 0.3 GPM limit specified for Turkey Point Unit No. 4.

d. Observed through-wall cracks in the dented regions, i.e., tube/support plate intersections, are constrained by the support plates; therefore, cracks should not burst during postulated accidents, unless the crack extends substantially beyond the tube support plate region.

e. Through wall cracks at dented locations, with the amount of leakages experienced to date (less than 0.3 GPM), have been stable during normal operation (no rapid failures), and are not anticipated to become unstable during postulated accidents.

f. Even though some non-through-wall cracks may exist and may crack through during postulated accidents, the associated leakage rate with such an event would be similar to that resulting from through wall crack found during normal operation and the crack would not be unstable. This consideration is consistent with the rationale upon which the preventive plugging limits were set for wastage or fretting type of degradation.

g. Even if a LOCA or a MSLB were to occur during the proposed period of operation and that some tubes were in a state of incipient failure, the consequence of such an event will not be severe as discussed in the safety evaluation included in the February 8, 1977 Order, pages 13 through 15.

4. The fact that most of inner row tubes are plugged lessens the concern

over the possible loss of lateral support of tubes due to the so-called "islanding" effect.

5. With respect to the tubes in the patch plate regions, no discernable flaws were found. As a preventive measure, tubes adjacent to plug welds in circulation holes were plugged. Therefore, the concern over the possible "new" hardspots around the patch plates is alleviated.

6. The residual phosphates in the sludge has caused continued wastage, but at a reduced rate, on the OD surfaces of tubes near the top of the tube sheet. The number of defective tubes (i.e., those degradations that exceeded the plugging limit) has been substantially reduced, in comparison with the previous inspections in 1975 and 1976; confirming that the extent of tube wastage is diminishing.

7. Since the extensive search conducted in the reactor lower plenum and other credible hide-out places in the RCS did not locate any of the missing plugs nor found any evidence of damage to the reactor internal components, we believe that the alleged eleven (11) missing tube plugs were never installed. In view of the occurrence at one PWR unit, however, there is a high probability that one or two plugs had or could become loose and deposit in the reactor lower plenum. In the event there are one or two loose plugs in the reactor vessel, they can be detected by the metal impact monitor that has been installed on site. The safety consequence of having one or two loose parts can then be evaluated.

We have also reviewed the proposed new procedures to augment the Quality Control Program, and have concluded that the photographic accounting procedures are adequate and, coupled with the installation of the on-site loose parts monitoring system, is acceptable.

Because of the need to assure that any stress corrosion cracking which occurs during operation remains small and stable, and that an extensive number of tubes do not incur penetrating cracks or substantial part thru-wall cracks, the staff had developed certain additional operating limitations that were specified in the February 8, 1977 NRC Order for the operation of Turkey Point Unit No. 4. These limitations, are designed to assure the detection of the onset of tube degradation before it becomes a significant concern. The limits on allowable primary to secondary leakage should be continued in effect during the next 6 months operating period.

Operational Limitations

1. A limit for primary to secondary leakage of 0.3 GPM will assure that no individual cracks will reach such proportions that it may become unstable during normal or accident loading conditions.

2. A substantial increase in the frequency at which leaking tubes are encountered could signal the development of more extensive general degradation. The potential for such a development during operation has been substantially alleviated by the limitations described below, requiring operation to be terminated in the event that the frequency of the detection of leaking tubes per plant should increase substantially to more than 1 in twenty days. Specifically, the restriction is that operation is to be terminated if two (2) or more tube leaks per plant occur during any twenty (20) day period. This restriction limits the potential number of heat up and cool down cycles resulting from tube plugging, and minimizes concern for possible thermal ratcheting.

3. At the end of the proposed six (6) month operating period, the unit shall be brought to cold shutdown condition for a re-inspection of the conditions of the steam generators and to re-assess the subsequent duration and mode of operation. The inspection program should be as comprehensive as the one performed during the recent outage. Detailed requirements will be determined by the NRC staff on the basis of the continuing operating experience.

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, under the conditions discussed above, and (2) such activities will be conducted in compliance with the Commission's regulation and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public. At the end of the next six equivalent months of operation, the condition of the steam generators will require further evaluation.

APPENDIX C

TURKEY POINT UNIT NO. 4

OPERATION WITH POTENTIAL LOOSE PART IN REACTOR SYSTEM

Introduction

As discussed in the licensee's submittals of July 27, 1977, July 29, 1977 and in Appendix B of this Safety Evaluation, the licensee made an extensive search of the primary coolant system to check for the possible presence of eleven "missing" steam generator plugs. During the current shutdown, the licensee removed the entire core and used television cameras in an inspection of the reactor vessel internals and primary coolant piping. The licensee also radiographed the smaller diameter piping.

Discussion

As a result of the licensee's inspection, the licensee concluded that there was no evidence of loose plugs in the reactor vessel or piping and that the "missing" plugs were probably never installed. To assure that this conclusion was valid, the licensee had installed a Metal Impact Monitoring System on the reactor vessel. During startup of the reactor coolant pumps on July 23, 1977, some noise was heard during the first pump runs. After the second eight minute pump runs, there was no detectable noise. Subsequently, the vendor, Westinghouse, made an analysis and concluded as described in the licensee's submittal of July 29, 1977 that: (1) a single object is located in the bottom of the reactor vessel, (2) the object is metallic with a mass between 0.1 and 1.0 pounds, and (3) the

object is now dormant in the reactor vessel, most likely at rest in a low flow area, within the bottom of the reactor vessel.

The licensee and the vendor have evaluated operation with this object in the reactor vessel and have concluded that it will not result in any adverse structural effects on the reactor vessel, its cladding, or reactor internals. They also concluded that no adverse thermal-hydraulic or other core conditions will result from the operation of the unit. We concur in these conclusions.

The licensee has commited to operate the Metal Impact Monitoring System (MIMS) until the next reactor vessel inspection. Any abnormal indications from the MIMS will be reported to the NRC. Also, following each startup from below 350° F, core barrel movement will be evaluated using neutron noise techniques.

Conclusion

The staff has evaluated the licensees actions and evaluation and concurs with his conclusion that operation with the metallic object in the reactor vessel will not result in any significant adverse thermal-hydraulic or structural effects.