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

April 25, 2001

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

SUBJECT: TURKEY POINT UNITS 3 AND 4 - ISSUANCE OF AMENDMENTS REGARDING EXTENSION OF RESIDUAL HEAT REMOVAL PUMP ALLOWED OUTAGE TIME FROM 72 HOURS TO 7 DAYS (TAC NOS. MB0429 AND MB0430)

Dear Mr. Plunkett:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No.212 to Facility Operating License No. DPR-31 and Amendment No. 206 to Facility Operating License No. DPR-41 for the Turkey Point Plant, Units Nos. 3 and 4, respectively. The amendments consist of changes to the Technical Specifications in response to your application dated October 30, 2000, as supplemented February 28, 2001.

These amendments would revise Technical Specification 5.3.2 for Turkey Point Units 3 and 4 to extend the residual heat removal (RHR) pump allowed outage time (AOT) from 72 hours to 7 days to restore an inoperable RHR pump to operable status. The extension of the AOT to 7 days is based on the projected time for the replacement of a leaking or failed pump shaft seal, the performance of post-maintenance testing, and the completion of any additional corrective actions that may be needed to restore the pump to operable status.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

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

Enclosures:
1. Amendment No.212 to DPR-31
2. Amendment No.206 to DPR-41
3. Safety Evaluation

cc w/encls: See next page

NRR-058

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WASHINGTON

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-250

TURKEY POINT PLANT UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 212
License No. DPR-31

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Florida Power and Light Company (the licensee) dated October 30, 2000, as supplemented February 28, 2001, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-31 is hereby amended to read as follows:

   (B) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 212 , are hereby incorporated in the license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

2. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard P. Correia

Richard P. Correia, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 25, 2001

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-251

TURKEY POINT PLANT UNIT NO. 4

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 206
License No. DPR-41

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Florida Power and Light Company (the licensee) dated October 30, 2000, as supplemented February 28, 2001, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-41 is hereby amended to read as follows:

   (B) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 206, are hereby incorporated in the license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard P. Correia

Richard P. Correia, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 25, 2001

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 212 FACILITY OPERATING LICENSE NO. DPR-31

AMENDMENT NO. 206 FACILITY OPERATING LICENSE NO. DPR-41

DOCKET NOS. 50-250 AND 50-251

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

Remove pages	Insert pages
3/4 5-3	3/4 5-3
3/4 5-4	3/4 5-4

EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 The following Emergency Core Cooling System (ECCS) equipment and flow paths shall be OPERABLE:

a. Four OPERABLE Safety Injection (SI) pumps, each capable of being powered from its associated OPERABLE diesel generator#, with discharge aligned to the RCS cold legs,*

b. Two OPERABLE RHR heat exchangers,

c. Two OPERABLE RHR pumps with discharge aligned to the RCS cold legs,

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank as defined in Specification 3.5.4, and

e. Two OPERABLE flow paths capable of taking suction from the containment sump.

APPLICABILITY: MODES 1, 2, and 3**.

ACTION:

a. With any one of the required ECCS components or flow paths inoperable, except for inoperable Safety Injection Pump(s) or an inoperable RHR pump, restore the inoperable component or flow path to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water in the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date since January 1, 1990.

c. With one of the four required Safety Injection pumps inoperable and the opposite unit in MODE 1, 2, or 3, restore the pump to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours.***

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*Only three OPERABLE Safety Injection (SI) pumps (two associated with the unit and one from the opposite unit), each capable of being powered from its associated OPERABLE diesel generator#, with discharge aligned to the RCS cold leg are required if the opposite unit is in MODE 4, 5, or 6.

**The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the Safety Injection flow paths isolated pursuant to Specification 3.4.9.3 provided that the Safety Injection flow paths are restored to OPERABLE status prior to Tavg exceeding 380°F. Safety Injection flow paths may be isolated when Tavg is less than 380°F.

***The provisions of Specifications 3.0.4 and 4.0.4 are not applicable.

#Inoperability of the required EDG's does not constitute inoperability of the associated Safety Injection pumps.

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

d. With two of the four required Safety Injection pumps inoperable and the opposite unit in MODE 1, 2, or 3, restore one of the two inoperable pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours. This ACTION applies to both units simultaneously.

e. With one of the three required Safety Injection pumps inoperable and the opposite unit in MODE 4, 5, or 6, restore the pump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDCWN within the following 6 hours.

f. With a required Safety Injection pump OPERABLE but not capable of being powered from its associated diesel generator, restore the capability within 72** hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

g. With an ECCS subsystem inoperable due to an RHR pump being inoperable, restore the inoperable RHR pump to OPERABLE status within 7 days or be in as least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

** 7 days for a Unit 3 diesel generator if the loss of capability is associated with replacement of the engine radiators prior to April 2000.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 212 TO FACILITY OPERATING LICENSE NO. DPR-31

AND AMENDMENT NO. 206 TO FACILITY OPERATING LICENSE NO. DPR-41

FLORIDA POWER AND LIGHT COMPANY

TURKEY POINT UNIT NOS. 3 AND 4

DOCKET NOS. 50-250 AND 50-251

1.0 INTRODUCTION

By application dated October 30, 2000, as supplemented February 28, 2001, Florida Power and Light Company (FPL, the licensee) requested changes to the technical specifications (TS) for the Turkey Point Plant, Units 3 and 4. The proposed amendments would allow extension of the allowed outage time (AOT) for an inoperable residual heat removal (RHR) pump from 72 hours to 7 days. This would allow greater flexibility in the scheduling and implementation of maintenance on the pump and avoid potential unscheduled plant shutdowns or requests for temporary relief for non-risk-significant conditions. The proposed extension is based on the projected time required to replace a leaking or failed pump shaft seal, perform post-maintenance testing, and complete any additional corrective actions that may be needed to restore the pump to operable status. The extended RHR pump AOT will provide time so that future seal repair activities can be completed successfully in a safe manner.

The licensee's supplementary submittal dated February 28, 2001, did not affect the original proposed no significant hazards determination, or expand the scope of the request as noticed in the *Federal Register* on December 27, 2000 (65 FR 81922).

2.0 BACKGROUND

2.1 Use of Probabilistic Risk Assessment in Evaluating TS Changes

Since the mid-1980s, the U. S. Nuclear Regulatory Commission (NRC) has been reviewing and granting improvements to TS that are based, at least in part, on probabilistic risk assessment (PRA) insights. In its final policy statement on TS improvements dated July 22, 1993, the NRC stated that it...

> ...expects that licensees, in preparing their Technical Specification related submittals, will utilize any plant-specific PSA (probabilistic safety assessment)[1]

[1]PSA and PRA are used interchangeably throughout this evaluation.

> or risk survey and any available literature on risk insights and PSAs. . . . Similarly, the NRC staff will also employ risk insights and PSAs in evaluating Technical Specifications related submittals. Further, as part of the Commission's ongoing program of improving Technical Specifications, it will continue to consider methods to make better use of risk and reliability information for defining future generic Technical Specifications requirements.

The NRC reiterated this point when it issued the revision to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.36, "Technical Specifications," in July 1995. In August 1995, the NRC adopted a final policy statement on the use of PRA methods in nuclear regulatory activities that encouraged greater use of PRA to improve safety decision-making and regulatory efficiency. The PRA policy statement included the following points:

- The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

- PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism associated with current regulatory requirements.

- PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.

2.2 Description of RHR System

Each of the two reactors at Turkey Point has a separate RHR system. The system is designed to provide low-pressure safety injection (LPSI) during accident conditions and decay heat removal during normal cooldowns and refueling evolutions. The system is aligned to support the LPSI function during normal plant operation. The system is designed such that two trains of equipment provide the post-accident flow delivery function under the most limiting single active failure condition. An RHR system consists of two trains, with each train consisting of a pump and a heat exchanger. The system utilizes high volume, low head centrifugal pumps to accomplish the flow delivery function.

The RHR pumps take suction from the refueling water storage tank (RWST) during the injection phase of a loss-of-coolant accident (LOCA), and pump water through a common discharge header. After entering containment, the RHR header splits into two paths with individual injection valves. After the injection valves, the RHR header combines with the high-head safety injection (HHSI) and accumulator discharge piping and directs flow through a common injection header into each of the three reactor coolant system (RCS) cold legs. The RHR pumps start and valves open upon receipt of a safety injection signal. The flow delivered by the RHR pumps supplements that provided by the safety injection accumulators in reflooding the reactor vessel to maintain core cooling during the early stages of a medium to large-break LOCA.

When the contents of the RWST are emptied to the RCS and the containment building, the plant operator takes action to transfer the suction of the emergency core cooling system to the containment recirculation sumps to permit recirculation of the injected fluid. During the recirculation phase of the recovery process, the RHR pumps take suction from the containment recirculation sumps and either direct fluid through the RHR heat exchangers back to the RCS or direct fluid through the RHR heat exchangers to the suction of the HHSI pump(s) and the containment spray pumps, and then back to the RCS and containment.

In addition to the LPSI function, the RHR system is used to remove core decay heat during normal cooldowns and long-term shutdowns. This additional cooling function is necessary because the steam generators do not provide an effective heat sink for the RCS at low temperatures. In the shutdown cooling alignment, the RHR pumps take suction from the RCS through a single hot leg penetration. A common suction line is used to route the coolant from the RCS to the suction of the RHR pumps. The common line splits in two at the suction of the pumps to form two separate RHR trains. The two trains reconverge downstream of the heat exchangers to form a common discharge header. A single pneumatic control valve is provided in the common discharge header to regulate the amount of flow that passes through the heat exchangers. This control function enables the reactor operator to control the RCS cooldown rate. A second pneumatic control valve is provided in a common bypass line around the heat exchangers. This valve compensates for the changes in the heat exchanger flow rate that will occur during a cooldown, to maintain a constant design flow rate to the core.

3.0 EVALUATION

The staff evaluated the licensee's proposed amendment to extend the TS completion time (completion time and AOT are used interchangeably herein) for one RHR pump out of service when in Modes 1, 2, and 3, from 72 hours to 7 days using insights derived from traditional engineering considerations and the use of PRA methods to determine the safety impact of extending the completion times.

3.1 Traditional Engineering Evaluation

The Turkey Point units are designed to be able to mitigate a design-basis LOCA assuming the worst single failure, including simultaneous loss-of-offsite power. For most sequences the failure of an emergency diesel generator is the limiting failure because it fails one train of the emergency core cooling system (including an RHR pump) completely. The units are unable to mitigate a large, medium, or small LOCA unless at least one RHR pump is available. For large- and medium-break LOCAs, the RHR pumps are needed to reflood the core following the LOCA. For small-break LOCAs, an RHR pump would be needed to act as a booster pump to the LPSI and HHSI pumps during the recirculation phase of injection.

Both RHR pumps are required to be operable by TS in Modes 1, 2, and 3. An inoperable RHR pump is a single failure and necessitates entry into an Action Statement (presently 72 hours), to limit the duration of continued plant operation with a degraded system. This time constitutes a temporary relaxation of the single failure criterion which, consistent with overall system reliability considerations, provides a limited time to fix equipment or otherwise make it operable. PRAs indicate that chances are small that the RHR pumps will be called upon in Modes 1, 2, or 3. The estimated frequency of a large-break LOCA is on the order of 10^{-5} per year. The estimated frequency of a small-break LOCA is on the order of 10^{-3} per year. The RHR system would also be used for RCS heat removal in the event of a steam generator tube rupture or other non-LOCA design-basis events, which have estimated frequencies on the order of 10^{-3} per year and lower. In contrast, an RHR pump will almost always be required in Modes 4, 5, and 6 since at least one RHR train is required to be in operation for RCS heat removal during normal shutdown operations and another RHR train is almost always required to be operable when in these modes. The scope of these license amendments is limited to the AOTs for the RHR pumps in Modes 1, 2, and 3.

The licensee based the duration of the proposed AOT on the anticipated time required to replace a leaking or failed pump shaft seal, perform post-maintenance testing, and complete any additional corrective actions that may be needed to restore the pump to operable status. Replacing an RHR pump seal is a very labor intensive evolution and requires that the entire pump (i.e., motor, shaft, and impeller) be unbolted and separated from the pump casing, removed from the auxiliary building, and placed on a temporary motor stand in the cask wash area for transport to the repair facility in the radwaste building. While the licensee indicates that past seal replacements have been performed within the current AOT of 72 hours, these have been primarily accomplished through intense focus of resources, and do not permit additional time for any re-work in the process, (i.e., there is no extra time for recovery prior to entering the shutdown portion of the AOT). Allowing for contingencies, extending the out-of-service time an additional 96 hours provides a high probability that future seal repairs will be successful while being performed in a safe manner.

In certain other cases, corrective maintenance and subsequent testing of an RHR pump and/or associated valves may require an RHR train to be out of service for more than a few days. In such cases, repair within the existing AOT cannot be assured and may result in an unscheduled plant shutdown or a request for NRC enforcement discretion to allow continued plant operation. A 7-day AOT would provide sufficient margin to effect most anticipated preventive and corrective maintenance activities and RHR system surveillance tests at power.

3.2 Probabilistic Risk Assessment Evaluation

The staff used a three-tiered approach to evaluate the risk associated with the proposed TS changes. The first tier evaluated the PRA model and the impact of the completion time extensions for an RHR pump on plant operational risk. The second tier addressed the need to preclude potentially high risk configurations, by identifying the need for any additional constraints or compensatory actions that, if implemented, would avoid or reduce the probability of a risk-significant configuration during the time when one RHR pump is out of service. The third tier evaluated the licensee's Configuration Risk Management Program (CRMP) to ensure that the applicable plant configuration will be appropriately assessed from a risk perspective before entering into or during the proposed completion times. Each tier and the associated findings are discussed below:

3.2.1 Tier 1 Evaluation

The licensee used traditional PRA methodology to evaluate the requested AOT extension for an RHR pump. The Tier 1 NRC staff review of the licensee's PRA involved three aspects:

(i) evaluation of the PRA model and application to the proposed AOT extension, (ii) evaluation of PRA results and insights stemming from the application, and (iii) discussion of the quality of the PRA.

(i) Evaluation of PRA Model and Application to the AOT Extension

The staff's review focused on the capability of the licensee's PRA model to analyze the risk stemming from the proposed AOT changes for RHR pumps, and did not involve an in-depth review of the licensee's PRA. This review was based on the staff's initial screening process, where the staff examined the licensee's internal events PRA results and recent operational experience regarding availability and reliability of RHR pumps. The staff concludes that the licensee's PRA results are reasonable, and the scope and depth of the PRA analysis support such a finding. Recent data for RHR and LPSI reliability and availability do not indicate any adverse trends.

The licensee's PRA includes both a Level 1 and Level 2 analysis. The analysis modeled both generic and plant-specific initiators, including internal flooding and dependencies that exist between initiating events and the associated mitigation systems. These initiators are consistent with those identified in previous PRAs. The licensee used both generic and plant-specific data. Since its response to Generic Letter (GL) 88-20, *Individual Plant Examination for Severe Accident Vulnerabilities*, and associated supplements, the licensee indicates its Reliability and Risk Assessment Group (RRAG) has maintained the PRA models consistent with the current plant configuration such that the licensee considers them "living" models.

(ii) Evaluation of PRA Results and Insights

The current estimated plant core damage frequency (CDF) is 9.0×10^{-6} per year, which is on the low end compared to that for other pressurized-water reactors. Sequences where LPSI or RHR pumps were used for mitigation tended to be at power sequences with initiators involving loss of inventory to the RCS (e.g., small-break or medium-break LOCAs).

The following baseline CDF and large early release frequency (LERF) were calculated by the licensee with the most current PRA model:

CDF = 9.0×10^{-6} per year
LERF = 3.8×10^{-8} per year

The licensee estimates the annual average CDF would increase to about 9.1×10^{-6} per year if the proposed 7-day AOT extension were granted. This is an increase of less than 1 percent, which is within the guidelines of Regulatory Guide (RG) 1.174. The licensee estimates the annual average LERF with the proposed 7-day AOT to be 3.8×10^{-8} per year, an increase of less than 1 percent also. The increases for CDF and LERF were less than 1 percent regardless of whether a best estimate or licensee-defined "upper bound" estimate were used.

The licensee provided the following values for the incremental conditional core damage probability (ICCDP) (excluding internal fire and external events):

ICCDP for the Corrective Maintenance case = 1.0×10^{-7} per year.
ICCDP for the Preventative Maintenance case = 3.2×10^{-8} per year.

These ICCDP values are below the staff guideline of 5.0 x10^{-7} per year from RG 1.177. The incremental conditional large early release probability was calculated by the licensee to be negligibly small, and also within the guidelines published in RG 1.177.

The PRA models used by the licensee to estimate the risk of the proposed AOT extension do not include a numerical estimate of the potential risk due to internal fires and external events. The licensee points out that in its response to GL 88-20, Supplement 4, "*Individual Plant Examination of External Events for Severe Accident Vulnerabilities*" (IPEEE), it concluded there were no severe accident vulnerabilities due to internal fires or external events. While the staff's review of the Turkey Point 3 and 4 IPEEE submittal concluded that the IPEEE process followed by the licensee was capable of identifying internal fire and external event vulnerabilities, the absence of perceived vulnerabilities (which was an undefined term in GL 88-20) is not in and of itself adequate justification for the acceptance of a proposed AOT extension. However, the RHR pumps, HHSI pumps, and charging pumps are not only in separate fire zones, but also in separate fire areas (i.e., there is little chance that a single fire would incapacitate some combination of these three sets of pumps). In addition, the licensee states it believes that "any potential impact the proposed RHR AOT extensions might have on the risk due to internal fires and external events would be very small and remain well below the acceptance criteria as stated in Reference Reg. Guide 1.177." The staff is satisfied that the separation provided by having the pumps in separate fire zones and areas, in conjunction with Appendix R requirements, provides adequate justification that the risk is low that a fire could fail multiple emergency core cooling system and RHR trains. Internal flooding was found to be a minimal contributor to risk due to the Turkey Point plant design.

(iii) Quality of the Turkey Point PSA

The models used for this application were generated by the licensee using the individual plant examination (IPE) models developed in response to GL 88-20, *Individual Plant Examination for Severe Accident Vulnerabilities*, and associated supplements. The original development work was classified and performed as "Quality Related" under the FPL 10 CFR Part 50, Appendix B quality assurance program. The revision and applications of the PSA models and associated databases continue to be handled as quality-related.

The licensee has indicated that administrative controls over its PSA program include written procedures, independent review of all model changes, data updates, and risk assessments performed using PSA methods and models. Risk assessments are performed by a PSA engineer and are independently reviewed. The licensee stated that, since the approval of the IPE, the RRAG at FPL has maintained the PSA models consistent with the current plant configuration such that they are considered "living" models. All computer programs that process PSA model inputs are verified and validated as needed. Software verification is the process used to ensure the software meets the software requirement specifications. Validation of software is performed by the licensee for different conditions such as a new installation of software, unreasonable results, or a change in computer configuration (software, hardware).

Multiple levels of review were used by the licensee in developing the Turkey Point PSA. The first consisted of normal engineering quality assurance practices carried out by the organization performing the analysis. The second level of review was performed by individuals from operations, technical staff, training, and the Independent Safety Engineering Group, who were not directly involved with the development of the PSA model. This provided diverse expertise with plant design and operations knowledge to review the system descriptions for accuracy. The third level of review was performed by PSA experts from ERIN Engineering. This review provided broad insights on techniques and results based on experience from other plant PSAs. The review team reviewed the PRA development procedures, as well as the output products. The licensee indicated that comments obtained from all the review sources were incorporated, as appropriate, into the work packages and the final product.

The Turkey Point IPE submittal to the NRC dated June 25, 1991, was reviewed extensively by the NRC and NRC contractors. In fact, the Turkey Point IPE was one of the few IPE submittals to receive a "Step 1" and a "Step 2" review by the NRC. The "Step 2" review consisted of a team of NRC representatives and contractors visiting FPL to conduct a week-long, extensive review of the Turkey Point IPE. Following these reviews, the Turkey Point IPE was revised in early 1992, and FPL received the NRC Safety Evaluation (SE) for the Turkey Point IPE on October 15, 1992. The SE concluded that the Turkey Point IPE had met the intent of GL 88-20.

The licensee stated that, prior to performing the risk assessment for this proposed license amendment, all design changes implemented since the last PSA update were reviewed. The licensee determined that changes to the PSA were not required. The licensee's submittal discussed the significant model changes incorporated since the IPE submittal.

3.2.2 Tier 2 Evaluation

The second tier addressed the need to preclude potentially high risk configurations, by identifying the need for any additional constraints or compensatory actions that, if implemented, would avoid or reduce the probability of a risk-significant configuration during the time when one RHR pump is out of service. The licensee stated that, based on previous maintenance-related risk evaluations and calculations performed in support of pre-evaluated maintenance risk assessment matrices, it did not identify any additional constraints or compensatory actions that should be included with the proposed AOT extension in order to avoid planned high risk configurations. Assessments performed in accordance with provisions of the proposed CRMP should ensure that potentially risk significant configurations are identified prior to removing an RHR pump from service for pre-planned maintenance. Similarly, implementation of the CRMP should ensure the proper evaluation of risk significance of unexpected configurations resulting from unplanned maintenance or conditions while within the AOT for an RHR pump.

3.2.3 Tier 3 Evaluation

Tier 3 is the development of a proceduralized program to ensure that the risk impact of out-of-service equipment is appropriately evaluated prior to performing a maintenance activity. A viable program is one that can uncover risk-significant plant equipment outage configurations in a timely manner during normal plant operation. The need for this third tier stems from the difficulty of identifying all possible risk-significant configurations under Tier 2 that will be encountered over extended periods of plant operation.

The licensee has committed that, in compliance with Section (a)(4) of the Maintenance Rule, 10 CFR Part 50.65, a CRMP based on the model program described in RG 1.177 will be implemented at Turkey Point to establish a proceduralized PRA-informed process to ensure that the overall impact of plant maintenance on plant risk is properly evaluated. Implementation of the CRMP should enable appropriate actions to be taken or decisions to be made to minimize and control risk when performing on-line maintenance with a risk-informed completion time.

The CRMP will provide a proceduralized risk-informed assessment to manage the risk associated with equipment inoperability. The licensee has committed that the CRMP and its essential elements will be described in the Turkey Point Administrative Procedure that will implement Section (a)(4) of the Maintenance Rule pursuant to 10 CFR Part 50.65. The program applies to TS structures, systems, or components for which risk-informed AOT has been granted. The program will include the following:

a. Provisions for the control and implementation of a Level 1 at-power internal events PSA-informed methodology. The assessment is to be capable of evaluating the applicable plant configuration.

b. Provisions for performing an assessment prior to entering the plant configuration described by the limiting conditions for operation (LCO) action statement for pre-planned activities.

c. Provisions for performing an assessment after entering the plant configuration described by the LCO action statement for unplanned entry into the LCO action statement.

d. Provisions for assessing the need for additional actions after the discovery of additional equipment out-of-service conditions while in the plant configuration described by the LCO Action Statement.

e. Provisions for considering other applicable risk-significant contributors such as Level 2 issues and external events, qualitatively or quantitatively.

3.3 Summary

The staff has evaluated the licensee's proposed changes for compliance with regulatory requirements as documented in this evaluation and has determined that they are acceptable. This determination is based on the following:

* The proposed RHR AOT modifications have only a minimal quantitative effect on plant risk. The calculated ICCDP for a single RHR AOT is small.

* The licensee's evaluation did not identify any additional constraints or compensatory actions that should be included with the proposed AOT extension in order to avoid planned high risk configurations.

* The licensee has proposed a risk-informed plant CRMP to assess the risk associated with the removal of equipment from service during the extended RHR pump AOT. The program provides the necessary assurances that appropriate assessments of plant risk configurations are sufficient to support the extended AOT request for the RHR pumps.

Therefore, the staff finds that the AOT for one RHR pump may be extended to 7 days, with a negligible impact on risk.

Based on the above evaluation, the staff concludes that the proposed TS changes are acceptable.

4.0 STATE CONSULTATION

Based upon a letter dated March 8, 1991, from Mary E. Clark of the State of Florida, Department of Health and Rehabilitative Services, to Deborah A. Miller, Licensing Assistant, NRC, the State of Florida does not desire notification of issuance of license amendments.

5.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (65 FR 81922). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: April 25, 2001

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TURKEY POINT PLANT

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