FPL

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L-2000-124
10 CFR 50.90

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

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Proposed License Amendments
Risk-Informed Residual Heat Removal Pump
Allowed Outage Time Extension

In accordance with 10 CFR 50.90, Florida Power and Light Company (FPL) requests that Appendix A of Facility Operating Licenses DPR-31 and DPR-41 for Turkey Point Units 3 and 4 be amended 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 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 adequate time so that future seal repair activities are completed successfully in a safe manner.

The present requirement to restore an inoperable RHR pump to operable status within 72 hours is specified in Technical Specification (TS) 3.5.2. There is a possibility that an RHR pump failure occurring during on line operation may result in a need for the corrective maintenance on the pump to take longer than 72 hours. With the current AOT, this may result in an unscheduled plant shutdown or a request for NRC enforcement discretion to allow continued plant operation while repairs are completed. To avoid these situations, a longer AOT is requested. The proposed amendments will revise TS 3.5.2 to allow up to 7 days to restore an inoperable RHR pump to operable status.

This is a risk-informed AOT extension request based on the results of a deterministic and probabilistic safety assessment performed by FPL using the Turkey Point Probabilistic Safety Assessment (PSA) and the methodology specified in Regulatory Guide (RG) 1.177. FPL concluded that the AOT change has only a small quantitative impact on plant risk, which meets the RG 1.177 acceptance guidelines for AOT changes.

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FPL has determined that the proposed license amendments do not involve a significant hazards consideration pursuant to 10 CFR 50.92. A description and justification of the amendments request is provided in Attachment 1. The no significant hazards consideration determination in support of the proposed Technical Specifications changes is provided in Attachment 2. Attachment 3 provides the proposed revised Technical Specifications pages. Attachment 4 contains the proposed revised Technical Specifications Bases pages for information only.

The proposed license amendments have been reviewed by the Turkey Point Plant Nuclear Safety Committee and the FPL Company Nuclear Review Board. In accordance with 10 CFR 50.91(b)(1), a copy of these proposed license amendments is being forwarded to the State Designee for the State of Florida. FPL requests that the proposed amendments, if approved, be issued by March, 2001.

Should there be any questions on this request, please contact us.

Very truly yours,

R. J. Hovey
Vice President
Turkey Point Plant

CLM

Attachments

cc: Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, Turkey Point Plant
Florida Department of Health and Rehabilitative Services

Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Proposed License Amendments
Risk Informed Residual Heat Removal Pump
Allowed Outage Time Extension

STATE OF FLORIDA)
)ss.
COUNTY OF MIAMI-DADE)

R. J. Hovey being first duly sworn, deposes and says:

That he is Vice President, Turkey Point Plant, of Florida Power and Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this document are true and correct to the best of his knowledge, information and belief, and that he is authorized to execute the document on behalf of said Licensee.

R. J. Hovey

Subscribed and sworn to before me this
30th day of October, 2000.

CHERYL A. STEVENSON
NOTARY PUBLIC - STATE OF FLORIDA
COMMISSION # CC929876
EXPIRES 6/19/2004
BONDED THRU ASA 1-888-NOTARY1

Cheryl A. Stevenson
Name of Notary Public (Type or Print)

R. J. Hovey is personally known to me.

ATTACHMENT 1

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1.0 Introduction

FPL requests that Appendix A of Facility Operating Licenses DPR-31 and DPR-41 for Turkey Point Units 3 and 4, respectively, be amended 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 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 are completed successfully in a safe manner.

The present requirement to restore an inoperable RHR pump to operable status within 72 hours is specified in Technical Specification (TS) 3.5.2. There is a possibility that an RHR pump failure occurring during on line operation may result in a need for the corrective maintenance on the pump to take longer than 72 hours. With the current AOT, this may result in an unscheduled plant shutdown or a request for NRC enforcement discretion to allow continued plant operation while repairs are completed. To avoid these situations, a longer AOT is requested. The proposed amendments will revise TS 3.5.2 to allow up to 7 days to restore an inoperable RHR pump to operable status.

This 7 day AOT extension request is based on the results of a deterministic and probabilistic safety assessment performed by FPL using the Turkey Point Probabilistic Safety Assessment (PSA) and the methodology specified in Regulatory Guide (RG) 1.177. FPL concluded that the AOT change has only a small quantitative impact on plant risk, which meets the RG 1.177 acceptance guidelines for AOT changes.

2.0 Background

2.1 System Description

Each reactor at Turkey Point has a separate RHR system. The system is designed to provide low pressure safety injection during accident conditions and decay heat removal during normal cooldowns and refueling evolutions.

The system is aligned to support the low pressure safety injection function during normal plant operation. Two trains of equipment are provided to ensure that the post-accident flow delivery function can be accomplished under the most limiting single active failure condition. 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 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 ECCS 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 (CS) pumps, and then back to the RCS and containment.

In addition to the low pressure safety injection 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 line splits in two at the suction of the pumps to form two separate RHR trains.

Each RHR train consists of a pump and a heat exchanger. The two trains re-converge to form a common discharge header downstream of the heat exchangers. 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.

2.2 Current Technical Specification Requirements

Technical Specification 3.5.2 addresses operability of the RHR pumps as a portion of the Emergency Core Cooling System (ECCS). In general, it requires redundant ECCS components to be operable with the plant in MODES 1 through 3. With any one ECCS component inoperable (except for a HHSI pump), the component must be restored to operable status within 72 hours or the reactor must be placed in HOT STANDBY within the next 6 hours, and be in HOT SHUTDOWN within the following 6 hours.

2.3 Previous Correspondence

A similar risk-informed AOT extension for an inoperable low presure safety injection train was granted for St. Lucie Units 1 and 2, by NRC letter dated February 15, 2000. The amendment was granted following St. Lucie's response to a request for additional information regarding the adequacy of Probabilistic Risk Assessments to support the requested TS changes. That information, as appropriate to Turkey Point, is included in this submittal (section 4.3.1).

3.0 Description of Proposed TS Changes:

The following changes to Technical Specification Action Statement 3.5.2 are proposed:

a) Change Action Statement a. of Technical Specification 3.5.2 to read as follows (proposed text addition shown in **bold**):

> "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) Add new Action Statement g. to Technical Specification 3.5.2 to read as follows (proposed addition in **bold**):

> **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 at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.**

The proposed changes are shown in Attachment 3.

4.0 Justification for Proposed TS Changes

4.1 Design Basis Requirements and Safety Analysis Impact

The design basis requirements for the RHR system are listed in Volume 4 of the Turkey Point Design Basis Documents. They are summarized as follows:

> The RHR System shall provide the low-head safety injection stage of the emergency core cooling function during a LOCA. The RHR System, in conjunction with the Safety Injection System(SIS), shall cool and recirculate the water that is collected in the containment recirculation sumps and return it to the RCS, SI system, and CS system to maintain reactor core and containment cooling functions during the cold leg recirculation phase following a LOCA. The RHR System shall be capable of delivering safety injection flow to the RCS, in the remote event that a passive failure occurs in the long term. (This type of passive failure is beyond the standard design basis passive failure of a 50-gpm leak.) The RHR System, in conjunction with the SIS, shall be capable of providing safety injection flow to the RCS hot legs during the long-term recirculation phase of a LOCA. The RHR System shall be capable of providing RCS mixing during operations involving a change in boron concentration in the RCS when the Reactor Coolant Pumps are not in operation. This function assures a homogeneous boron concentration throughout the RCS during either a boration or dilution event.

Both RHR pumps are required to be operable by Technical Specifications. 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 consititutes 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. The amendments proposed herein lengthen that limited time, bounded by considerations of probable need (section 4.2) and risk significance (section 4.3).

4.2 Deterministic Assessment of the Proposed RHR Pump AOT Extension

In Modes 1, 2, and 3, the RHR trains must be operable in the event that LOCA mitigation becomes necessary. The estimated frequency of a large LOCA is on the order of E-05 per year. The estimated frequency of a small LOCA is on the order of E-03 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 E-03 per year and lower. In contrast, at least one RHR train is required to be operable for RCS heat removal during normal shutdown operations in Modes 4, 5, and 6, and is almost always in operation when in these modes. Therefore, performing preventive and corrective maintenance on RHR trains when at power can enhance overall plant safety by increasing the availability and

reliability of the RHR system for normal shutdown cooling operations, i.e., when it is most often needed.

The duration of the proposed AOT is based 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 a 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 past seal replacements have been performed within the current AOT of 72 hours, these have been primarily accomplished through intense focus of resources, and does not permit any 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 a RHR pump and/or associated valves may require a 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.

4.3 Probabilistic Safety Assessment of the Proposed RHR Pump AOT Extension

The risk assessment of the proposed AOT extension for Turkey Point was generated using an updated version of the IPE model developed in response to Generic Letter (GL) 88-20, *"Individual Plant Examination for Severe Accident Vulnerabilities"*, and associated supplements. Since submittal of the IPE, both the model and the reliability/unavailability databases for Turkey Point have been updated. The two units are sufficiently similar such that one model represents both units. The updated model and databases were used to calculate the risk numbers to evaluate the extended RHR AOT. The model update process included a review of all plant design changes that were implemented since creation of the original models. A summary of the Turkey Point PSA changes since submittal of the IPE is included in section 4.3.1.4.2 of this attachment.

FPL's evaluation of the risk associated with the proposed AOT conforms to the three-tiered approach that is identified in Regulatory Position C.2.3 of USNRC Regulatory Guide 1.177, *An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications*, August 1998 (RG 1.177). Tier 1 consists of the PSA capability and insights; Tier 2 identifies risk-significant plant configurations that should be avoided; and Tier 3 describes the implementation of a risk-informed configuration risk management program. The acceptability of the increase in risk is measured against the "small increase" guideline of RG 1.177, and against the "very small increase" guideline of Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,* July 1998 (RG 1.174).

4.3.1 Tier 1, Analysis of Risk Impact and Calculated Results

Tier 1 is an evaluation of the impact on plant risk of the proposed TS change as expressed by the change in core damage frequency (CDF), the incremental conditional change in core damage probability (ICCDP), and when appropriate, the change in large early release frequency (LERF) and the incremental conditional large early release probability (ICLERP). The definitions of these last two risk measures, as given in RG 1.177, are shown below:

ICCDP = [(conditional CDF with the subject equipment out of service) - (baseline CDF with nominal expected equipment unavailabilities)] * (duration of single AOT under consideration)

ICLERP = [(conditional LERF with the subject equipment out of service) – (baseline with nominal expected equipment unavailabilities)] * (duration of single AOT under consideration)

Two cases were evaluated for the overall change in core damage frequency: a best estimate case, and an upper bound case. For the best estimate case, the number of hours of RHR Preventive Maintenance activities (PMs) currently performed during shutdown conditions which would be performed on line if the AOT extension is granted was estimated, and added to the existing RHR train maintenance unavailabilities. For the upper bound case, the RHR train maintenance unavailabilities were increased to reflect an additional 7 days outage time per year per train. The core damage frequencies and large early release frequencies associated with these two cases were calculated and are shown below in Tables 1 and 2.

Table 1

CDFs and LERFs for Best Estimate Case
Post-AOT Extension

Risk Measure	Baseline	Best Estimate	Absolute Increase over Baseline	% Increase over Baseline
CDF	9.011E-06	9.016E-06	5.000E-09	0.06%
LERF	3.784E-08	3.786E-08	2.100E-11	0.06%

Table 2

CDF and LERFs for Upper Bound Case
Post-AOT Extension

Risk Measure	Baseline	Upper Bound	Absolute Increase over Baseline	% Increase over Baseline
CDF	9.011E-06	9.089E-06	7.800E-08	0.86%
LERF	3.784E-08	3.791E-08	7.325E-11	0.19%

It can be seen from the data in Tables 1 and 2 that the calculated increases in CDF are less than 1E-06 per reactor year for both the best estimate and upper bound case. It can also be seen that the calculated increases in LERF for both cases are less than 1E-07 per reactor year. Thus, the RG 1.174 acceptance guideline of "very small" increases in these parameters is satisfied.

In addition to the CDF and LERF calculations, FPL calculated the ICCDP (Table 3) and ICLERP (Table 4) corresponding to the requested 7 day AOT for comparison to acceptance guidelines defined in (a) RG 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis*, and (b) RG 1.177. A sensitivity case was calculated for the ICLERP where the early containment failure fraction of non-bypass core damage frequency was increased from the nominal value of 0.076% to 1%. The results are included in Table 4.

Table 3

ICCDP RESULTS
(Calculated using RG 1.177 methodology)

Description	ICCDP
ICCDP for Corrective Maintenance (CM) case	1.024E-07
ICCDP for Preventive Maintenance (PM) case	3.239E-08

Table 4

ICLERP RESULTS
(Calculated using RG 1.177 methodology)

Parameter	Early Containment Failure = 0.076%	Early Containment Failure = 1%*
ICLERP for Corrective Maintenance (CM) case	7.769E-11	1.020E-09
ICLERP for Preventive Maintenance (PM) case	2.480E-11	3.223E-10

* Sensitivity evaluation

The calculated ICCDP for the corrective maintenance case (Table 3) is 1.024E-07, less than the RG 1.177 guideline definition of a "small" quantitative impact on plant risk as an ICCDP of 5E-07 or less. The calculated ICCDP for the preventive maintenance case (Table 3) is 3.239E-08, also less than the RG 1.177 guideline definition of a "small" quantitative impact on plant risk. The calculated ICLERPs for both cases (Table 4) are less than 5E-08, satisfying the RG 1.177 guideline definition of a "small" quantitative impact on plant risk as an ICLERP of 5E-08 or less.

4.3.1.1 Modeling Adequacy and Completeness Relative to this Application. The results of the evaluations performed in support of the Turkey Point proposed RHR AOT extension were reviewed by two PSA engineers (a preparer and an independent reviewer) from FPL's Nuclear Engineering Reliability and Risk Assessment Group. Both concluded that the results were appropriate considering the inputs and assumptions used, and based on a review of the dominant cutsets, that the results are reasonable and the models are adequate for this application. The following summarizes the dominant cutsets:

Attachment 1-A lists the top 10 baseline cutsets. This CDF is reflected in Tables 1 and 2 as the Baseline CDF. The dominant cutsets are small-small LOCA sequences with failures related to high pressure safety injection/recirculation, and a medium LOCA sequence with failure of the operator to successfully switch to cold leg recirculation. Other cutsets in the top 10 include station blackout, loss of all feedwater and failure of feed-and-bleed cooling, and ATWS.

Attachment 1-B lists the top 10 cutsets for the corrective maintenance (CM) case. This CDF is used along with the baseline CDF in the calculation of the ICCDP for the CM case (see Table 3). For this case, one RHR train is assumed out of service for corrective maintenance, and the common cause RHR pump failure probability is set to the beta factor probability. The top cutset is that of a medium LOCA followed by a common cause failure of the RHR pumps. The sixth and seventh cutsets are medium LOCAs followed by failure of the 'B' RHR train with the 'A' RHR train out of service for corrective maintenance. The remainder of the cutsets consist of the higher frequency cutsets from the baseline case.

Attachment 1-C lists the top 10 cutsets for the preventive maintenance (PM) case. This CDF is used along with the baseline CDF in the calculation of the ICCDP for the PM case (see Table 4). For this case, one RHR train is assumed out-of-service for preventive maintenance and the common cause RHR failures are set to 0.0. These cutsets are very similar to those in Attachment 1-B, with the exception of the absence of the cutset of medium LOCA with a common cause failure of the RHR pumps.

Attachment 1-D lists the top 10 cutsets for the new average CDF using the best estimate of RHR downtime post-AOT extension. The CDF is reflected in Table 1 as the Best Estimate CDF. For this case, the RHR train maintenance unavailabilities were changed based on the best estimate of proposed downtime assuming an increased AOT. The dominant sequences are the same as the baseline case.

Attachment 1-E lists the top 10 cutsets for the new average CDF using an upper bound estimate of RHR downtime post-AOT extension. The CDF is reflected in Table 2 as the Upper Bound CDF. For this case, the RHR train maintenance unavailabilities were changed based on an upper bound estimate of the proposed downtime assuming an increased AOT. The dominant sequences are the same as the baseline case.

4.3.1.2 Internal Fires and External Events. The PSA models used to calculate the estimated risk impact of the proposed AOT extension do not include an assessment of the potential risk due to internal fires and external events. The Turkey Point response to GL 88-20, Supplement 4, "*Individual Plant Examination of External Events for Severe Accident Vulnerabilities*" (IPEEE), concluded that there were no severe accident vulnerabilities due to internal fires and external events. FPL believes that any potential impact the AOT extension might have on the risk due to internal fires and external events would be very small and remain well below the acceptance criteria.

The required action in response to external events is well proceduralized. The following is a summary of applicable plant procedures that address plant actions in response to external events, e.g., hurricanes, tornadoes, and fires:

The procedure entitled "Duties of Emergency Coordinator" provides the criteria for emergency classification of any natural phenomena event. It includes criteria for emergency classification of events related to hurricane warnings, hurricanes, tornadoes, flood surge, earthquakes, and fires.

The procedure entitled "Natural Emergencies" provides instructions and guidelines for preparing, controlling and recovering the plant following activation of the Emergency Plan for natural emergencies. This procedure is used when the natural emergency meets the criteria specificed in "Duties of Emergency Coordinator." It addresses tornadoes, hurricanes and earthquakes, but is to be used for any severe natural disturbance which results in Emergency Plan activation. Specific guidance is provided for staffing in preparation for a hurricane, and for coping with possible flood conditions associated with more intense hurricanes. It provides the criteria for unit shutdown if a hurricane warning is in effect, and either or both units are operating. The shutdown criteria are as follows:

- For storms projected to reach Category 1 or 2, the unit(s) shall be placed in HOT STANDBY (Mode 3) or below at least two (2) hours before the projected onset of sustained hurricane force winds at the site, and both units shall remain off-line for the duration of the hurricane force winds (or restoration of reliable offsite power).

- For storms projected to reach Category 3, 4 and 5 prior to landfall, the unit(s) shall be shut down, maintaining RCS temperature between 343°F and 350°F Tave. and steam generator pressure greater than 85 psig. RHR should be placed in service and AFW should be aligned and operable. These plant conditions shall be established at least two (2) hours before the projected onset of sustained hurricane force winds at the site and both units shall remain off-line for the duration of the hurricane force winds (or restoration of reliable offsite power).

The Off Normal Operating procedure entitled "Severe Weather Preparations" provides instructions to be followed to prepare for severe weather (including tornadoes) or in response to a hurricane watch or warning. Actions to be taken include, but are not limited to:

- Topping off the diesel oil storage tanks,
- Performing flood protection stoplog inspection to verify operability and adequate inventory of flood protection equipment (RHR pumps are below grade),
- Performing test runs of the emergency diesel generators (and other diesels on site),
- Installing any removed hatches on the Auxiliary Building roof

There are Off-Normal Operating procedures which provide operator actions for responding to a fire at Turkey Point. These procedures provide specific guidance to the operator for performing a safe shutdown fire impact assessment and direction as to which mode to place the unit in if the fire challenges continued unit operation or stable plant conditions. Additional procedures provide fire-fighting strategies to assist the fire brigade in combating a fire.

4.3.1.3 Consideration of Cumulative Impact of Risk-Informed AOTs. FPL is preparing to submit a license amendment for a risk-informed AOT extension (from 3 days to 14 days) for the Turkey Point EDG trains. Accordingly, the cumulative impact on the average CDF of both the proposed EDG and RHR AOT changes was evaluated and found to be approximately 1.5E-07 (Table 5). The average change in LERF is 1.2E-10 (Table 5). Both of these values are within Region III of RG 1.174 Figures 3 and 4, respectively, and are thus considered very small.

Table 5

CDFs and LERFs for Best Estimate Case
Post-EDG and RHR AOT Extension

Risk Measure	Baseline	Best Estimate	Absolute Increase over Baseline	% Increase over Baseline
CDF	9.013E-06	9.164E-06	1.510E-07	1.68%
LERF	3.785E-08	3.797E-08	1.213E-10	0.32%

4.3.1.4 Quality of the Turkey Point PSA. The models used for this application were generated using the IPE models developed in response to Generic Letter (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 50, Appendix B quality assurance (QA) program. The revision and applications of the PSA models and associated databases continue to be handled as Quality Related.

Administrative controls 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, independently reviewed by another PSA engineer, and approved by the Department Head or designee. The Reliability and Risk Assessment Group (RRAG) is required to follow the FPL Nuclear Engineering Quality Instructions (QI) with written procedures derived from those QIs. Procedures, risk assessment documentation, and associated records are controlled and retained as QA records.

Since the approval of the IPE, the RRAG has maintained the PSA models consistent with the current plant configuration such that they are considered "living" models. The PSA models are updated for different reasons, including plant changes and modifications, procedure changes, accrual of new plant data, discovery of modeling errors, advances in PSA technology, and issuance of new industry PSA standards. The update process ensures that the applicable changes are implemented and documented in a timely manner so that risk analyses performed in support of plant operations reflect the current plant configuration, operating philosophy, and transient and component failure history. The PSA maintenance and update process is described in the RRAG Standard entitled, *Probability Safety Assessment Update and Maintenance Procedure*. This standard defines two types of periodic updates: 1) a data analysis update, and 2) a model update. The data analysis update is performed at least every five years. Model updates consist of either single or multiple PSA changes and are performed at a frequency dependent on the estimated impact of the accumulated changes. Guidelines to determine the need for a model update are provided in the standard.

4.3.1.4.1 PSA Software. All computer programs that process PSA model inputs are verified and validated as needed. The RRAG policy on verification and validation of QA controlled/procured software, as well as the verification and validation for software and computers when used for Quality Related applications are described in the RRAG Standard entitled, *Probability Safety Assessment Software Control Procedure*. This standard provides a list of all the software used by the RRAG and indicates whether the software is QA controlled/procured. Software verification is the process used to ensure the software meets the software requirement specifications. The PSA software that is procured with a QA option and is developed under a 10 CFR 50, Appendix B, QA program does not require further software verification by the RRAG. However, the PSA software, which is not procured with a QA option can be verified by comparison of results to previously approved software.

Validation of software is performed for different conditions such as: 1) a new installation of software, 2) any new database or configuration file changes issued by the RRAG, 3) unreasonable results, 4) change in computer configuration (software, hardware), or 5) use of software for Quality Related applications for the first time. Validation requirements for each Quality Related PSA computer program are documented in a Software Verification/Validation Plan (SVVP) procedure. These requirements include the method of validation, the frequency of validation, the documentation required and the acceptance criteria. A SVVP procedure is submitted for each program. Actual validation benchmark problems can exercise more than one program, but a separate Software Verification/Validation Report (SVVR) must be submitted for each program. Each SVVP procedure and SVVR is independently reviewed and then approved by the RRAG supervisor. Software validation tests both the software and the hardware. Validation tests are also performed following any significant change in the hardware, operating system, or program, or if the validation period established in the SVVP procedure expires.

4.3.1.4.2 Model Changes Since Submittal of the IPE. Prior to performing the risk assessment for this proposed license amendment, all design changes implemented since the last PSA update were reviewed. Changes to the PSA were not required as a result of this review.

A summary of significant model changes incorporated since the IPE submittal follows:

The replacement of one of the standby steam generator feedwater pumps with a diesel-driven pump, and the removal of the black-start diesel generators were incorporated into the model. Minor improvements were made in the modeling of instrument air, chemical and volume control, HVAC, AC power, component cooling water, and service water systems.

The success criteria for small LOCAs was modified to take credit for cooldown, depressurization, and use of the opposite unit's RWST inventory for injection. The RCP seal LOCA treatment was modified to reflect the latest research in this area.

A complete data update was performed, including all plant-specific failure rates, test and maintenance unavailabilities, initiating event frequencies, and common-cause beta factors. New initiating event (IE) frequencies were calculated for all LOCAs. This was done in accordance with CEOG Probabilistic Safety Assessment Working Group (PSAWG) Technical Position Paper, *"Evaluation of the Initiating Event Frequency for the Loss of Coolant Accident"*, CEOG Task 941, January 1997. Although the IE frequencies for the larger LOCA sizes decreased, the net impact was an increase in the total LOCA IE frequency of nearly 40%.

The process of adding recoveries is now automated using a recovery "rule file." The rule file utilizes a manual recovery action process in that recovery actions are added to each cutset rather than being generated from the model, but the process is automated such that all the similar cutset scenarios are recovered automatically. This automatic feature ensures uniform and complete inclusion of recovery actions throughout all of the generated cutsets, and yields more realistic and consistent results. The methodology for crediting the recovery of offsite power was changed to a more realistic convolution analysis technique.

4.3.1.4.3 PSA Reviews. As discussed in the Turkey Point IPE submittal, multiple levels of review were used for the Turkey Point PSA. The first consisted of normal engineering quality assurance practices carried out by the organization performing the analysis. A qualified individual with knowledge of PSA methods and plant systems performed an independent review of the results for each task. This represents a detailed check of the input to the PSA model and provides a high degree of quality assurance.

The second level of review was performed by plant personnel not directly involved with the development of the PSA model. This review was performed by individuals from Operations, Technical Staff, Training, and the Independent Safety Engineering Group (ISEG), who reviewed the system description notebooks and accident sequence description. 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. Comments obtained from all the review sources were incorporated, as appropriate, into the work packages, and the final product.

Following the Turkey Point IPE submittal to the NRC on June 25, 1991, it 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 Report (SER) for the Turkey Point IPE on October 15, 1992. The SER concluded that the Turkey Point IPE had met the intent of Generic Letter 88-20.

4.3.2 Tier 2, Avoidance of Risk-Significant Plant Configurations

Tier 2 is the avoidance of risk-significant plant configurations by identification of potentially high risk configurations that could exist if equipment in addition to that associated with the TS change is concurrently taken out of service or other risk significant operational factors such as concurrent system or equipment testing are involved. The objective of Tier 2 is to ensure that appropriate restrictions are placed on dominant risk significant configurations that would be relevant to the proposed TS change. Based on previous maintenance related risk evaluations and calculations performed in support of Pre-Evaluated Maintenance Risk Assessment (PREMRA) matrices, FPL has not identified 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 will ensure that potentially risk significant configurations are identified prior to removing a RHR train from service for pre-planned maintenance. Similarly, implementation of the CRMP will ensure the proper evaluation of risk significance of unexpected configurations resulting from unplanned maintenance or conditions while in the risk-informed AOT.

4.3.3 Tier 3, Configuration Risk Management

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 would be one that is able to 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.

In compliance with Section (a)(4) of the Maintenance Rule, 10 CFR 50.65, a Configuration Risk Management Program (CRMP) based on the model program described in Regulatory Guide 1.177 will be implemented at Turkey Point to establish a proceduralized probabilistic risk assessment-informed process to ensure that the overall impact of plant maintenance on plant risk is properly evaluated. Implementation of the CRMP will 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 Configuration Risk Management Program will provide a proceduralized risk-informed assessment to manage the risk associated with equipment inoperability. The Configuration Risk Management Program 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 50.65. The program applies to technical specification structures, systems, or components (SSC) for which risk-informed allowed outage time 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.

5.0 Environmental Impact Evaluation

10 CFR 51.22(c)(9) provides criteria for identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not:

(i) involve a significant hazards consideration,

(ii) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and

(iii) result in a significant increase in individual or cumulative occupational radiation exposure.

FPL has reviewed these proposed license amendments and concludes that they 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 needs to be prepared in connection with this request.

6.0 Conclusion

The risk contributions associated with extending the AOT for a single inoperable RHR pump from 72 hours to 7 days have been quantitatively evaluated using the current plant-specific Probabilistic Safety Assessment for Turkey Point Units 3 and 4.

FPL has determined that defense-in-depth philosophy is maintained with the proposed AOT, and has evaluated the risk-impact from potential RHR pump unavailability using the three-tiered approach for performing risk assessments that is identified in regulatory guidelines. The calculated increases in the average CDF and LERF are "very small" as defined in RG 1.174. The calculations performed for ICCDP and ICLERP demonstrate that the ICCDP and ICLERP values are less than the RG 1.177 definition of "small" for both the corrective maintenance case and the preventive maintenance case. The Tier 3 CRMP should further serve to reduce any increase in risk associated with the RHR AOT extension.

Implementation of the extended AOT will likely result in higher reliability and availability of the RHR system during shutdown conditions, where it is often the main source of core cooling. In addition, the RHR AOT extension may serve to avoid a plant shutdown, thereby likely reducing the overall risk.

Relative to the average core damage frequency calculated for the appropriate severe accidents, NUREG/CR-6141 states, "A risk-based AOT assures that the single event and yearly AOT risk contributions are acceptable." The proposed 7 day RHR AOT qualifies as a beneficial risk-informed AOT, and FPL believes that the proposed amendments are acceptable.

Cutsets for Baseline CDF

#	Inputs	Description	Event Probability	Cutset Frequency
1	%ZZMU3	MEDIUM LOCA	5.18E-05	1.55E-06
	U3OPMLPR	OPERATOR FAILS TO SWITCHOVER TO COLD LEG RECIRC (MEDIUM LOCA)	3.00E-02	
2	%ZZS1U3	SMALL-SMALL LOCA S1	2.85E-03	9.98E-07
	GMMNC3843I	COMMON CAUSE FAILURE OF MOV-3-843A, B	3.50E-04	
3	%ZZS1U3	SMALL-SMALL LOCA S1	2.85E-03	4.83E-07
	GMM4A215I	CCF FOR 4/4 HHSI PUMPS FAIL TO START	1.69E-04	
4	%ZZT7U3	SPURIOUS UNIT 3 SAFETY INJECTION SIGNAL	1.50E-01	2.70E-07
	N30002	FAILURE OF CONTROL RODS TO INSERT WITH POWER REMOVED	1.80E-06	
5	%ZZS1U3	SMALL-SMALL LOCA S1	2.85E-03	2.34E-07
	ITM0400013	FAN V8A UNAVAILABLE DUE TO TEST OR MAINTENANCE	8.20E-03	
	U0RABFAN	OPERATOR FAILS TO START ONE RAB FAN	1.00E-02	
6	%ZZS1U3	SMALL-SMALL LOCA S1	2.85E-03	2.12E-07
	GXVK3867	MANUAL VALVE 3-867 TRANSFERS CLOSED	7.45E-05	
7	%ZZT7U3	SPURIOUS UNIT 3 SAFETY INJECTION SIGNAL	1.50E-01	1.95E-07
	NMM3CCFRT	TRIP BREAKER FAILS TO OPEN DUE TO COMMON CAUSE	1.30E-05	
	X3OPKMRODI	OPERATOR FAILS MANUAL ROD INSERTION WITHIN 1 MIN.	1.00E-01	
8	%ZZLOG	LOSS OF GRID	7.49E-02	1.75E-07
	EMM3ACLR	FAILURE OF 3A BUS BREAKERS TO CLEAR	4.13E-02	
	EMM3BCLR	FAILURE OF BUS 3B BREAKERS TO CLEAR	4.13E-02	
	XLOGCS1R	OSP NON-RECOVERY, CASE 1R	1.30E-01	
	ZZSL	RCP SEAL LOCA FLAG	2.10E-01	
	ZZXCROSST	FAILURE TO ALIGN BLACKOUT XTIE (OPERATOR AND HARDWARE)	5.01E-02	
9	%ZZLOG	LOSS OF GRID	7.49E-02	1.46E-07
	AHFL0N2BKU	OPERATOR LEAVES THE BACKUP N2 SYSTEM MISALIGNED	3.00E-03	
	FMM0P82B	MODULE FOR SSGFP B FAILS	7.44E-02	
	HMM3M331	LOCAL FAULTS IN HEADER M 331 (UNIT 3 STANDBY AIR COMPRESSOR)	1.52E-01	
	HMM4M431	LOCAL FAULTS HEADER M 431 (U4 RUNNING AIR COMP)	1.52E-01	
	XLOGCS5	OSP NON-RECOVERY, CASE 5	3.80E-01	
10	%ZZT1U3	REACTOR TRIP	1.04E+00	1.13E-07
	EDGF33A	DIESEL GENERATOR 3A FAILS TO RUN	8.38E-02	
	NMM3CCFRT	TRIP BREAKER FAILS TO OPEN DUE TO COMMON CAUSE	1.30E-05	
	X3OPKMRODI	OPERATOR FAILS MANUAL ROD INSERTION WITHIN 1 MIN.	1.00E-01	

CDF w/1 RHR Train Unavailable for ICCDP CM Case

#	Inputs	Description	Event Prob	Probability
1	%ZZMU3	MEDIUM LOCA	5.18E-05	3.32E-06
	JMMA3C210	COMMON CAUSE FAILURE OF RHR/LHSI PUMPS TO START	6.41E-02	
2	%ZZMU3	MEDIUM LOCA	5.18E-05	1.55E-06
	U3OPMLPR	OPERATOR FAILS TO SWITCHOVER TO COLD LEG RECIRC (MEDIUM LOCA)	3.00E-02	
3	%ZZS1U3	SMALL-SMALL LOCA S1	2.85E-03	9.98E-07
	GMMNC3843I	COMMON CAUSE FAILURE OF MOV-3-843A, B	3.50E-04	
4	%ZZMU3	MEDIUM LOCA	5.18E-05	4.88E-07
	JMM3H200I	FAILURE OF RHR PUMP TRAIN B DURING INJECTION PHASE	9.42E-03	
	JTMPMP3AI	RHR TRAIN 3A IN TEST OR MAINTENANCE	1.00E+00	
5	%ZZS1U3	SMALL-SMALL LOCA S1	2.85E-03	4.83E-07
	GMM4A215I	CCF FOR 4/4 HHSI PUMPS FAIL TO START	1.69E-04	
6	%ZZMU3	MEDIUM LOCA	5.18E-05	4.80E-07
	JMM3H700R	FAILURE OF RHR PUMP B TRAIN DURING RECIRCULATION PHASE	9.27E-03	
	JTMPMP3AR	RHR LHSI TRAIN 3A IN TEST OR MAINTENANCE	1.00E+00	
7	%ZZMU3	MEDIUM LOCA	5.18E-05	2.74E-07
	JMM3H703R	INDEPENDENT LOCAL FAULTS IN CCW LINE TO 3B RHR HX	5.30E-03	
	JTMPMP3AR	RHR LHSI TRAIN 3A IN TEST OR MAINTENANCE	1.00E+00	
8	%ZZT7U3	SPURIOUS UNIT 3 SAFETY INJECTION SIGNAL	1.50E-01	2.70E-07
	N30002	FAILURE OF CONTROL RODS TO INSERT WITH POWER REMOVED	1.80E-06	
9	%ZZS1U3	SMALL-SMALL LOCA S1	2.85E-03	2.34E-07
	ITM0400013	FAN V8A UNAVAILABLE DUE TO TEST OR MAINTENANCE	8.20E-03	
	U0RABFAN	OPERATOR FAILS TO START ONE RAB FAN	1.00E-02	
10	%ZZS1U3	SMALL-SMALL LOCA S1	2.85E-03	2.12E-07
	GXVK3867	MANUAL VALVE 3-867 TRANSFERS CLOSED	7.45E-05	

CDF w/1 RHR Train Unavailable for ICCDP PM Case

#	Inputs	Description	Event Prob	Probability
1	%ZZMU3	MEDIUM LOCA	5.18E-05	1.55E-06
	U3OPMLPR	OPERATOR FAILS TO SWITCHOVER TO COLD LEG RECIRC (MEDIUM LOCA)	3.00E-02	
2	%ZZS1U3	SMALL-SMALL LOCA S1	2.85E-03	9.98E-07
	GMMNC3843I	COMMON CAUSE FAILURE OF MOV-3-843A, B	3.50E-04	
3	%ZZMU3	MEDIUM LOCA	5.18E-05	4.88E-07
	JMM3H200I	FAILURE OF RHR PUMP TRAIN B DURING INJECTION PHASE	9.42E-03	
	JTMPMP3AI	RHR TRAIN 3A IN TEST OR MAINTENANCE	1.00E+00	
4	%ZZS1U3	SMALL-SMALL LOCA S1	2.85E-03	4.83E-07
	GMM4A215I	CCF FOR 4/4 HHSI PUMPS FAIL TO START	1.69E-04	
5	%ZZMU3	MEDIUM LOCA	5.18E-05	4.80E-07
	JMM3H700R	FAILURE OF RHR PUMP B TRAIN DURING RECIRCULATION PHASE	9.27E-03	
	JTMPMP3AR	RHR LHSI TRAIN 3A IN TEST OR MAINTENANCE	1.00E+00	
6	%ZZMU3	MEDIUM LOCA	5.18E-05	2.74E-07
	JMM3H703R	INDEPENDENT LOCAL FAULTS IN CCW LINE TO 3B RHR HX	5.30E-03	
	JTMPMP3AR	RHR LHSI TRAIN 3A IN TEST OR MAINTENANCE	1.00E+00	
7	%ZZT7U3	SPURIOUS UNIT 3 SAFETY INJECTION SIGNAL	1.50E-01	2.70E-07
	N30002	FAILURE OF CONTROL RODS TO INSERT WITH POWER REMOVED	1.80E-06	
8	%ZZS1U3	SMALL-SMALL LOCA S1	2.85E-03	2.34E-07
	ITM0400013	FAN V8A UNAVAILABLE DUE TO TEST OR MAINTENANCE	8.20E-03	
	U0RABFAN	OPERATOR FAILS TO START ONE RAB FAN	1.00E-02	
9	%ZZS1U3	SMALL-SMALL LOCA S1	2.85E-03	2.12E-07
	GXVK3867	MANUAL VALVE 3-867 TRANSFERS CLOSED	7.45E-05	
10	%ZZT7U3	SPURIOUS UNIT 3 SAFETY INJECTION SIGNAL	1.50E-01	1.95E-07
	NMM3CCFRT	TRIP BREAKER FAILS TO OPEN DUE TO COMMON CAUSE	1.30E-05	
	X3OPKMRODI	OPERATOR FAILS MANUAL ROD INSERTION WITHIN 1 MIN.	1.00E-01	

CDF w/Best Estimate RHR Post-AOT Extension Maintenance Unavailability Case

#	Inputs	Description	Event Probability	Cutset Frequency
1	%ZZMU3	MEDIUM LOCA	5.18E-05	1.55E-06
	U3OPMLPR	OPERATOR FAILS TO SWITCHOVER TO COLD LEG RECIRC (MEDIUM LOCA)	3.00E-02	
2	%ZZS1U3	SMALL-SMALL LOCA S1	2.85E-03	9.98E-07
	GMMNC3843I	COMMON CAUSE FAILURE OF MOV-3-843A, B	3.50E-04	
3	%ZZS1U3	SMALL-SMALL LOCA S1	2.85E-03	4.83E-07
	GMM4A215I	CCF FOR 4/4 HHSI PUMPS FAIL TO START	1.69E-04	
4	%ZZT7U3	SPURIOUS UNIT 3 SAFETY INJECTION SIGNAL	1.50E-01	2.70E-07
	N30002	FAILURE OF CONTROL RODS TO INSERT WITH POWER REMOVED	1.80E-06	
5	%ZZS1U3	SMALL-SMALL LOCA S1	2.85E-03	2.34E-07
	ITM0400013	FAN V8A UNAVAILABLE DUE TO TEST OR MAINTENANCE	8.20E-03	
	U0RABFAN	OPERATOR FAILS TO START ONE RAB FAN	1.00E-02	
6	%ZZS1U3	SMALL-SMALL LOCA S1	2.85E-03	2.12E-07
	GXVK3867	MANUAL VALVE 3-867 TRANSFERS CLOSED	7.45E-05	
7	%ZZT7U3	SPURIOUS UNIT 3 SAFETY INJECTION SIGNAL	1.50E-01	1.95E-07
	NMM3CCFRT	TRIP BREAKER FAILS TO OPEN DUE TO COMMON CAUSE	1.30E-05	
	X3OPKMRODI	OPERATOR FAILS MANUAL ROD INSERTION WITHIN 1 MIN.	1.00E-01	
8	%ZZLOG	LOSS OF GRID	7.49E-02	1.75E-07
	EMM3ACLR	FAILURE OF 3A BUS BREAKERS TO CLEAR	4.13E-02	
	EMM3BCLR	FAILURE OF BUS 3B BREAKERS TO CLEAR	4.13E-02	
	XLOGCS1R	OSP NON-RECOVERY, CASE 1R	1.30E-01	
	ZZSL	RCP SEAL LOCA FLAG	2.10E-01	
	ZZXCROSST	FAILURE TO ALIGN BLACKOUT XTIE (OPERATOR AND HARDWARE)	5.01E-02	
9	%ZZLOG	LOSS OF GRID	7.49E-02	1.46E-07
	AHFL0N2BKU	OPERATOR LEAVES THE BACKUP N2 SYSTEM MISALIGNED	3.00E-03	
	FMM0P82B	MODULE FOR SSGFP B FAILS	7.44E-02	
	HMM3M331	LOCAL FAULTS IN HEADER M 331 (UNIT 3 STANDBY AIR COMPRESSOR)	1.52E-01	
	HMM4M431	LOCAL FAULTS HEADER M 431 (U4 RUNNING AIR COMP)	1.52E-01	
	XLOGCS5	OSP NON-RECOVERY, CASE 5	3.80E-01	
10	%ZZT1U3	REACTOR TRIP	1.04E+00	1.13E-07
	EDGF33A	DIESEL GENERATOR 3A FAILS TO RUN	8.38E-02	
	NMM3CCFRT	TRIP BREAKER FAILS TO OPEN DUE TO COMMON CAUSE	1.30E-05	
	X3OPKMRODI	OPERATOR FAILS MANUAL ROD INSERTION WITHIN 1 MIN.	1.00E-01	

CDF w/Upper Bound RHR Post-AOT Extension Maintenance Unavailability Case

#	Inputs	Description	Event Probability	Cutset Frequency
1	%ZZMU3	MEDIUM LOCA	5.18E-05	1.55E-06
	U3OPMLPR	OPERATOR FAILS TO SWITCHOVER TO COLD LEG RECIRC (MEDIUM LOCA)	3.00E-02	
2	%ZZS1U3	SMALL-SMALL LOCA S1	2.85E-03	9.98E-07
	GMMNC3843I	COMMON CAUSE FAILURE OF MOV-3-843A, B	3.50E-04	
3	%ZZS1U3	SMALL-SMALL LOCA S1	2.85E-03	4.83E-07
	GMM4A215I	CCF FOR 4/4 HHSI PUMPS FAIL TO START	1.69E-04	
4	%ZZT7U3	SPURIOUS UNIT 3 SAFETY INJECTION SIGNAL	1.50E-01	2.70E-07
	N30002	FAILURE OF CONTROL RODS TO INSERT WITH POWER REMOVED	1.80E-06	
5	%ZZS1U3	SMALL-SMALL LOCA S1	2.85E-03	2.34E-07
	ITM0400013	FAN V8A UNAVAILABLE DUE TO TEST OR MAINTENANCE	8.20E-03	
	U0RABFAN	OPERATOR FAILS TO START ONE RAB FAN	1.00E-02	
6	%ZZS1U3	SMALL-SMALL LOCA S1	2.85E-03	2.12E-07
	GXVK3867	MANUAL VALVE 3-867 TRANSFERS CLOSED	7.45E-05	
7	%ZZT7U3	SPURIOUS UNIT 3 SAFETY INJECTION SIGNAL	1.50E-01	1.95E-07
	NMM3CCFRT	TRIP BREAKER FAILS TO OPEN DUE TO COMMON CAUSE	1.30E-05	
	X3OPKMRODI	OPERATOR FAILS MANUAL ROD INSERTION WITHIN 1 MIN.	1.00E-01	
8	%ZZLOG	LOSS OF GRID	7.49E-02	1.75E-07
	EMM3ACLR	FAILURE OF 3A BUS BREAKERS TO CLEAR	4.13E-02	
	EMM3BCLR	FAILURE OF BUS 3B BREAKERS TO CLEAR	4.13E-02	
	XLOGCS1R	OSP NON-RECOVERY, CASE 1R	1.30E-01	
	ZZSL	RCP SEAL LOCA FLAG	2.10E-01	
	ZZXCROSST	FAILURE TO ALIGN BLACKOUT XTIE (OPERATOR AND HARDWARE)	5.01E-02	
9	%ZZLOG	LOSS OF GRID	7.49E-02	1.46E-07
	AHFL0N2BKU	OPERATOR LEAVES THE BACKUP N2 SYSTEM MISALIGNED	3.00E-03	
	FMM0P82B	MODULE FOR SSGFP B FAILS	7.44E-02	
	HMM3M331	LOCAL FAULTS IN HEADER M 331 (UNIT 3 STANDBY AIR COMPRESSOR)	1.52E-01	
	HMM4M431	LOCAL FAULTS HEADER M 431 (U4 RUNNING AIR COMP)	1.52E-01	
	XLOGCS5	OSP NON-RECOVERY, CASE 5	3.80E-01	
10	%ZZT1U3	REACTOR TRIP	1.04E+00	1.13E-07
	EDGF33A	DIESEL GENERATOR 3A FAILS TO RUN	8.38E-02	
	NMM3CCFRT	TRIP BREAKER FAILS TO OPEN DUE TO COMMON CAUSE	1.30E-05	
	X3OPKMRODI	OPERATOR FAILS MANUAL ROD INSERTION WITHIN 1 MIN.	1.00E-01	

ATTACHMENT 2

No Significant Hazards Consideration Determination

Description of Proposed License Amendments

The purpose of the proposed license amendments (PLA) is 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 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 are completed successfully in a safe manner.

The present requirement to restore an inoperable RHR pump to operable status within 72 hours is specified in Technical Specification (TS) 3.5.2. The proposed amendments will revise TS 3.5.2 to allow up to 7 days to restore an inoperable RHR pump to operable status. This is a risk-informed AOT extension request based on the results of a deterministic and probabilistic safety assessment performed by FPL using the Turkey Point Probabilistic Safety Assessment (PSA). FPL concluded from the results of that assessment that the risk contribution of the AOT extension is very small, and that the net impact of the proposed amendment may be risk neutral.

Introduction

The Nuclear Regulatory Commission has provided standards for determining whether a proposed license amendment involves no significant safety hazards consideration (10 CFR §50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Each standard is discussed below for the proposed amendments.

Discussion

(1) **Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.**

The RHR system is part of the Emergency Core Cooling System. Inoperable RHR pumps are not accident initiators in any accident previously evaluated, and an extended AOT to restore operability of an inoperable RHR pump would not increase the probability of occurrence of accidents previously analyzed. Therefore, this change does not involve an increase in the probability of an accident previously evaluated.

The RHR system is primarily designed to mitigate the consequences of the large Loss Of Coolant Accident (LOCA). In addition, the RHR system provides for primary system heat removal during unit shutdown conditions. The proposed changes do not affect any of the assumptions relative to accident initiators or accident response provided in the plant safety analyses. Accordingly, the consequences of accidents previously evaluated do not change.

A Probabilistic Safety Assessment (PSA) was performed to evaluate the impact of extending the allowed outage time on the RHR pump from 72 hours to 7 days. FPL concluded from the results of that assessment that the risk contribution of the AOT extension is very small, and that the net impact of the proposed amendment may be risk neutral.

Therefore, the change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

(2) Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed change does not alter the design, physical configuration, or modes of operation of the plant. Plant configurations that are prohibited by Technical Specifications will not be created by the AOT extension. Therefore, the proposed activity does not create the possibility of a new or different kind of accident from any previously evaluated.

(3) Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The margin of safety associated with the Emergency Core Cooling System is established by acceptance criteria for system performance defined in 10 CFR 50.46. The proposed amendments will not change these acceptance criteria or the operability requirements for equipment that is used to achieve such performance as demonstrated in the plant safety analyses. Moreover, a Probabilistic Safety Assessment of the risk impact of extending the AOT for a single inoperable RHR pump has concluded that the risk contribution is very small, RHR system reliability can potentially be improved, and the net impact of the proposed change may be risk neutral.

Therefore, the change does not involve a significant reduction in a margin of safety.

Summary

Based on the above discussion, FPL has determined that the proposed amendments do not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety; and therefore the proposed changes do not involve a significant safety hazards consideration as defined in 10 CFR 50.92.

Environmental Impact Evaluation

10 CFR 51.22(c)(9) provides criteria for identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not:

(iv) involve a significant hazards consideration,

(v) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and

(vi) result in a significant increase in individual or cumulative occupational radiation exposure.

FPL has reviewed these proposed license amendments and concludes that they 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 needs to be prepared in connection with this request.

ATTACHMENT 3

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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**.

or an inoperable RHR pump

ACTION:

a. With any one of the required ECCS components or flow paths inoperable, except for inoperable Safety Injection Pump(s), 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 T_{avg} exceeding 380°F. Safety Injection flow paths may be isolated when T_{avg} 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 SHUTDOWN 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 at 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.

ATTACHMENT 4

For Information Only
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ECCS SUBSYSTEMS (Continued)

With the RCS temperature below 350°F, operation with less than full redundant equipment is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

TS Surveillance 4.5.2.a requires that each ECCS component and flow path be demonstrated operable at least once per 12 hours by verifying by control room indication that the valves listed in section 4.5.2.a are in the indicated positions with power to the valve operators removed. "Verifying control room indication" applies to the valve position and not to the valve operator power removal. The breaker position may be verified by either the off condition of the breaker position indication light in the Control Room, or the verification of the locked open breaker position in the field. Verifying that power is removed to the applicable valve operators can be accomplished by direct field indication of the breaker (locked in the open position), or by observation of the breaker position status lamp in the control room (lamp is off when breaker is open). Surveillance Requirements for throttle valve position stops prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration.

Pump performance requirements are obtained from accident analysis assumptions. Varying flowrates are provided to accommodate testing during modes and alignments.

In the RHR test, differential head is specified in "feet." This criteria will allow for compensation of test data with water density due to varying temperature.

3/4.5.4 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The indicated water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The temperature limits on the RWST solution ensure that: 1) the solubility of the borated water will be maintained, and 2) the temperature of the RWST solution is consistent with the LOCA analysis. Portable instrumentation may be used to monitor the RWST temperature.

TS 3.5.2, Action g. provides an allowed outage/action completion time (AOT) of up to 7 days to restore an inoperable RHR pump to OPERABLE status, provided the affected ECCS subsystem is inoperable only because its associated RHR pump is inoperable. This 7 day AOT is based on the results of a deterministic and probabilistic safety assessment, and is referred to as a "risk-informed" AOT extension. Planned entry into this AOT requires that a risk assessment be performed in accordance with the Configuration Risk Management Program (CRMP), which is described in the administrative procedure that implements the maintenance rule pursuant to 10CFR50.65.