5.0 Environmental Impacts of Postulated Accidents

Environmental issues associated with postulated accidents were discussed in the *Generic Environmental Impact Statement for License Renewal of Nuclear Plants* (GEIS), NUREG-1437, Volumes 1 and 2 (NRC 1996a; 1999).[a] The GEIS included a determination of whether the analysis of the environmental issue could be applied to all plants and whether additional mitigation measures would be warranted. Issues were then assigned a Category 1 or a Category 2 designation. As set forth in the GEIS, Category 1 issues are those that meet all of the following criteria:

(1) The environmental impacts associated with the issue have been determined to apply either to all plants or, for some issues, to plants having a specific type of cooling system or other specified plant or site characteristics.

(2) Single significance level (i.e., SMALL, MODERATE, or LARGE) has been assigned to the impacts (except for collective offsite radiological impacts from the fuel cycle and from high-level waste and spent fuel disposal).

(3) Mitigation of adverse impacts associated with the issue has been considered in the analysis, and it has been determined that additional plant-specific mitigation measures are not likely to be sufficiently beneficial to warrant implementation.

For issues that meet the three Category 1 criteria, no additional plant-specific analysis is required unless new and significant information is identified.

Category 2 issues are those that do not meet one or more of the criteria for Category 1, and therefore, additional plant-specific review of these issues is required.

This chapter describes the environmental impacts from postulated accidents that might occur during the license renewal term.

5.1 Postulated Plant Accidents

Two classes of accidents are evaluated in the GEIS. These are design-basis accidents (DBAs) and severe accidents, as discussed below.

(a) The GEIS was originally issued in 1996. Addendum 1 to the GEIS was issued in 1999. Hereafter, all references to the "GEIS" include the GEIS and Addendum 1.

5.1.1 Design-Basis Accidents

In order to receive U.S. Nuclear Regulatory Commission (NRC) approval to operate a nuclear power facility, an applicant must submit a Safety Analysis Report (SAR) as part of its application. The SAR presents the design criteria and design information for the proposed reactor and comprehensive data on the proposed site. The SAR also discusses various hypothetical accident situations and the safety features that are provided to prevent and mitigate accidents. The NRC staff reviews the application to determine whether the plant design meets the Commission's regulations and requirements and includes, in part, the nuclear plant design and its anticipated response to an accident.

DBAs are those accidents that both the licensee and the NRC staff evaluate to ensure that the plant can withstand normal and abnormal transients, and a broad spectrum of postulated accidents without undue hazard to the health and safety of the public. A number of these postulated accidents are not expected to occur during the life of the plant, but are evaluated to establish the design basis for the preventive and mitigative safety systems of the facility. The acceptance criteria for DBAs are described in 10 CFR Part 50 and 10 CFR Part 100. The environmental impacts of DBAs are evaluated during the initial license process, and the ability of the plant to withstand these accidents is demonstrated to be acceptable before issuance of the operating license (OL). The results of these evaluations are found in license documentation such as the staff's Safety Evaluation Report (SER), the Final Environmental Statement (FES), the licensee's Final Safety Analysis Report (FSAR), and Section 5.1 of this SEIS. The licensee is required to maintain the acceptable design and performance criteria throughout the life of the plant including any extended-life operation. The consequences for these events are evaluated for the hypothetical maximum exposed individual; as such, changes in the plant environment will not affect these evaluations. Because of the requirements that continuous acceptability of the consequences and aging management programs be in effect for license renewal, the environmental impacts as calculated for DBAs should not differ significantly from initial licensing assessments over the life of the plant, including the license renewal period. Accordingly, the design of the plant relative to DBAs during the extended period is considered to remain acceptable and the environmental impacts of those accidents were not examined further in the GEIS.

The Commission has determined that the environmental impacts of DBAs are of SMALL significance for all plants because the plants were designed to successfully withstand these accidents. Therefore, for the purposes of license renewal, design-basis events are designated as a Category 1 issue in 10 CFR Part 51, Subpart A, Appendix B, Table B-1. The early resolution of the DBAs makes them a part of the current licensing basis of the plant; the current licensing basis of the plant is to be maintained by the licensee under its current license and, therefore, under the provisions of 10 CFR 54.30, is not subject to review under license renewal. This issue, applicable to Turkey Point Units 3 and 4, is listed in Table 5-1. Florida Power &

Table 5-1. Category 1 Issue Applicable to Postulated Accidents During the Renewal Term

ISSUE—10 CFR Part 51, Subpart A, Appendix B, Table B-1	GEIS Section
POSTULATED ACCIDENTS	
Design-basis accidents (DBAs)	5.3.2; 5.5.1

Light (FPL) stated in its Environmental Report (ER; FPL 2000) that it is not aware of any new and significant information associated with the renewal of the Turkey Point Units 3 and 4 OLs. The staff has not identified any significant new information during its independent review of the FPL ER (FPL 2000), the staff's site visit, the scoping process, or its evaluation of other available information. Therefore, the staff concludes that there are no impacts related to this issue beyond those discussed in the GEIS.

5.1.2 Severe Accidents

Severe nuclear accidents are those that are more severe than DBAs because they could result in substantial damage to the reactor core, whether or not there are serious offsite consequences. The GEIS assessed the impacts of severe accidents during the license renewal period, using the results of existing analyses and site-specific information to conservatively predict the environmental impacts of severe accidents for each plant during the renewal period.

Based on information in the GEIS, the Commission found that

> "The probability weighted consequences of atmospheric releases, fallout onto open bodies of water, releases to ground water, and societal and economic impacts from severe accidents are small for all plants. However, alternatives to mitigate severe accidents must be considered for all plants that have not considered such alternatives."

The Commission has designated severe accidents as a Category 2 issue in 10 CFR Part 51, Subpart A, Appendix B, Table B-1. This issue, applicable to Turkey Point Units 3 and 4, is listed in Table 5-2.

The staff has not identified any significant new information with regard to the consequences from severe accidents during its independent review of the FPL ER (FPL 2000), the staff's site visit, the scoping process, or its evaluation of other available information. Therefore, the staff concludes that there are no impacts of severe accidents beyond those discussed in the GEIS. However, in accordance with 10 CFR 51.53(c)(3)(ii)(L), the staff has reviewed severe accident

Table 5-2. Category 2 Issue Applicable to Postulated Accidents During the Renewal Term

ISSUE—10 CFR Part 51, Subpart A, Appendix B, Table B-1	GEIS Sections	10 CFR 51.53(c)(3)(ii) Subparagraph	SEIS Section
POSTULATED ACCIDENTS			
Severe Accidents	5.3.3; 5.3.3.2; 5.3.3.3; 5.3.3.4; 5.3.3.5; 5.4; 5.5.2	L	5.2

mitigation alternatives (SAMAs) for Turkey Point Units 3 and 4. The results of its review are discussed in Section 5.2.

5.2 Severe Accident Mitigation Alternatives

Title 10 of the Code of Federal Regulations, Section 51.53(c)(3)(ii)(L), requires that license renewal applicants consider alternatives to mitigate severe accidents if the staff has not previously considered SAMAs for the applicant's plant in an environmental impact statement or related supplement or in an environmental assessment. The purpose of this consideration is to ensure that plant design and procedure changes with the potential for improving severe accident safety performance are identified and evaluated. SAMAs have not been previously considered for Turkey Point Units 3 and 4; therefore, the following sections address those alternatives.

5.2.1 Introduction

FPL submitted an assessment of SAMAs for Turkey Point Units 3 and 4 as part of the ER (FPL 2000). The assessment was based on the Turkey Point Probabilistic Safety Assessment (PSA) for total accident frequency (core damage frequency [CDF] and containment release frequency), and a supplemental analysis of offsite consequences and economic impacts for risk determination. While identifying and evaluating potential SAMAs, FPL took into consideration the insights and recommendations from several SAMA analyses for other plants, other NRC and industry documents discussing potential plant improvements, and documented insights provided by the plant staff. FPL considered 167 SAMAs and concluded that there are no SAMAs that are cost-beneficial associated with license renewal.

Based on a review of the SAMA assessment, the NRC issued a request for additional information (RAI) to FPL by letter dated January 31, 2001 (NRC 2001a). Key questions concerned the base case risk and its constituents, PSA model and changes, external events and their limited inclusion in SAMAs, and potential design enhancements and their disposition. FPL submitted additional information in response to the staff's RAIs by letter dated March 30, 2001 (FPL 2001). These responses addressed the staff's concerns and reaffirmed the conclusions of the study.

An assessment of SAMAs for Turkey Point Units 3 and 4 is presented below.

5.2.2 Estimate of Risk for Turkey Point Units 3 and 4

FPL's estimates of offsite risk at Turkey Point Units 3 and 4 are summarized below. The summary is followed by an evaluation of FPL's risk estimates.

5.2.2.1 FPL's Risk Estimates

Two distinct analyses are combined to form the basis for the risk estimates used in the SAMA analysis: (1) the Turkey Point PSA model, which is an updated version of the individual plant examination (IPE), and (2) a supplemental analysis of offsite consequences and economic impacts for risk determination developed specifically for SAMA analyses. The Turkey Point PSA is considered to be a living plant risk model, incorporating new information on equipment performance, plant configuration changes, and refinements in PSA modeling techniques. It contains a Level 1 analysis to determine the CDF from internally initiated events and a Level 2 analysis to determine containment performance during severe accidents. The baseline CDF for the purpose of SAMA evaluation is 1.62×10^{-5}/yr. A breakdown of the CDF is provided in Table 5-3. As shown in the table, transient initiators contribute about 39 percent, while loss-of-coolant accidents (LOCAs) contribute about 60 percent of the total internal events CDF. It is seen in Table 5-3 that containment bypass events (i.e., steam generator tube rupture [SGTR] and interfacing systems loss-of-coolant accident [ISLOCA]) make a minimal contribution to internal events CDF for Turkey Point, and the frequency associated with the largest release (i.e., ISLOCA) for Turkey Point is estimated to be about 6×10^{-8} per reactor year (ry). The station blackout (SBO) contribution to the transients is not explicitly provided in the submittal; however, the plant damage states for which both sprays and fan coolers have failed (mostly due to loss of power) is about 4.49×10^{-8}/ry. Anticipated transient without scram (ATWS) contributors are not explicitly provided in the submittal; however, based on the top 20 cutsets, ATWS contributes, at least, 1×10^{-6}/ry.

The offsite consequences and economic impact analyses use the MELCOR Accident Consequence Code System 2 (MACCS2) code, Version 1.12, to determine the offsite risk impacts on the surrounding environment and the public. Inputs for this analysis include plant/site-specific values for core radionuclide inventory, source term and release fractions, meteorological data,

Table 5-3. Turkey Point Core Damage Frequency (CDF)

Initiating Event	Frequency (per reactor year)
Transients	6.3×10^{-6}
Loss-of-coolant accident (LOCA)	9.8×10^{-6}
Steam generator tube rupture (SGTR)	1.7×10^{-8}
Interfacing system LOCA	6.2×10^{-8}
Total CDF from internal events	1.6×10^{-5}

projected population distribution, emergency response evacuation modeling, and economic data. The magnitude of the onsite impacts (in terms of clean-up and decontamination costs and occupational dose) is based on information provided in NUREG/BR-0184 (NRC 1997a).

FPL estimates the risk to the population within 80 km (50 mi) of the Turkey Point site, from internal initiators, to be 10.9 person-rem/yr. Table 5-4 shows the contributions to population dose by containment release mode. Late containment failure accounts for the majority of the population dose. This is primarily due to the dominance of the late containment failure frequency (i.e., about 9.05×10^{-6}/ry), which is about 56 percent of the total internal events CDF of 1.62×10^{-5}/ry, or 99 percent of the total release frequency of 9.14×10^{-6}/ry. (Note that about 44 percent of the postulated core melt scenarios at Turkey Point do not result in containment failure and the release of radioactivity.) The contribution of early containment failure, including containment bypass scenarios, is very small (about 0.5 percent of total internal events CDF or about 1 percent of total release frequency).

In response to an RAI, FPL (FPL 2001) explains that the dominant late containment failure sequences are due to the conservative assumptions made in the IPE/PSA with respect to exceeding the equipment qualification (EQ) limit for a short period of time causing the failure of

Table 5-4. Risk Profile

Containment Release Mode	Contribution to Population Dose (%)
Containment intact	0
Late containment failure	97.2
Early containment failure	0.1
Containment bypass	2.7

the containment heat removal systems (CHRSs). Plant damage states with successful containment spray but with hypothesized late containment failures are the result of noncondensible gas generation due to protracted core-concrete-interactions. Basemat melt-through (BMT) contributes about 25 percent (under dry and wet cavity conditions), and loss of containment integrity due to hydrogen burn contributes about 25 percent. FPL indicated (FPL 2001) that if the conservative assumptions (i.e., EQ-induced failure of CHRS and BMT, considering Severe Accident Management Guidelines [SAMGs]) were to be removed from the Level 2 analysis, the late containment failure contribution would be expected to drop from approximately 56 percent to 25 percent (due to hydrogen burn causing late containment failure).

5.2.2.2 Review of FPL's Risk Estimates

FPL's determination of offsite risk impacts at Turkey Point Units 3 and 4 is based on the Turkey Point PSA and a separate MACCS2 analysis. This review considered the following major elements:

- the Level 1 and 2 risk models
- the modifications to the PSA model
- the MACCS2 analyses performed to translate fission product release frequencies from the Level 2 PSA model into offsite consequence measures.

Each of these analyses was reviewed to determine the acceptability of FPL's risk estimates for the SAMA analysis, as summarized below.

The staff's review of the Turkey Point IPE is described in a staff report dated October 15, 1992 (NRC 1992). In that review, the staff evaluated the methodology, models, data, and assumptions used to estimate the CDF and characterize containment performance and fission product releases. The staff concluded that FPL's analysis met the intent of Generic Letter 88-20 (NRC 1988); that is, the IPE was of adequate quality to be used to look for design or operational vulnerabilities. Although the staff reviewed certain aspects of the IPE in more detail than others, it primarily focused on the licensee's ability to examine Turkey Point for severe accident vulnerabilities and not specifically on the detailed findings or quantification estimates. Overall, the staff believed that the Turkey Point IPE was of adequate quality to be used as a tool in searching for areas with high potential for risk reduction and to assess such risk reductions, especially when the risk models are used in conjunction with insights, such as those from risk importance, sensitivity, and uncertainty analyses. It is important to note that significant changes have been made to the Turkey Point risk model since the original IPE was completed and reviewed by the NRC staff. These include both modifications to the models and changes due to plant modification, as discussed below.

A comparison of CDF profiles between the original IPE and the current PSA indicates that the estimate of the CDF for internal events has been reduced from 3.7×10^{-4} /ry to about 1.62×10^{-5}/ry (over a factor of 20 reduction). The lower values in the current PSA are attributed to plant and modeling improvements that have been implemented in Turkey Point since the IPE, as discussed below.

The original model documented in the 1991 Turkey Point IPE submittal had a CDF of 3.7×10^{-4}/ry. To address NRC comments, the model was revised and submitted to the NRC in 1992. The Turkey Point PSA model was updated in 1993, 1995, and 1997 to incorporate plant and modeling changes. The CDF for the 1997 update was 6.12×10^{-5}/ry. Plant upgrades incorporated in the 1997 revised model included modifications to the service water system, standby steam generator feedwater pump (from motor-driven to diesel-driven) and instrument air system upgrade. Major modeling changes included time-dependent recovery of offsite power, more consistent recovery actions (use of rule-based recovery), and data updates. In 1999, the 1997 Turkey Point PSA model was modified to account for several plant features that have significant impact on the benefit calculations, but were not included in the plant risk model. The modified baseline CDF is 1.62×10^{-5}/ry. This CDF was used to evaluate SAMAs related to component cooling water (CCW) performance, reactor coolant pump (RCP) seal LOCA, secondary heat removal, and equipment ventilation, and takes credit for the following features:

- cross-tie of the Unit 3 and Unit 4 CCW systems reducing the loss of CCW initiator frequency and allowing recovery post-accident

- alternate feedwater sources for the steam generators, including cross-tie via the opposite unit main feedwater and condensate supply systems

- revised dependency on reactor auxiliary building (RAB) ventilation to reflect that only residual heat-removal pumps require RAB fans

- revised common cause start and run failure beta factors for high head safety injection (HHSI) pumps based on INEL-94/0064, Volume 6, and

- revised likelihood for RCP seal LOCA upon loss of seal cooling (partially due to the new O-ring for the RCPs).

The present CDF value of 1.62×10^{-5}/ry is lower than most of the original IPE values estimated for other pressurized water reactors (PWRs) with large dry containments, although many of these have similarly been reduced due to modeling and hardware changes since submitting their IPEs. Figure 11.6 of NUREG-1560 (NRC 1997b) shows that the IPE-based total internal events CDF for Westinghouse 3-loop plants ranges from 6×10^{-5} to 4×10^{-4}/ry.

As noted in Table 5-3, the CDFs for SGTR and ISLOCA were very low. In an RAI (NRC 2001) the staff requested an explanation of why these values were so low, when compared both with the original IPE values for Turkey Point Units 3 and 4 and with corresponding values for similar plants. According to the FPL response (FPL 2001), the CDF reduction for SGTR was primarily based on crediting the redundant and diverse secondary heat removal mechanisms. The SGTR Emergency Operating Procedure (EOP) provides detailed guidance on bringing the reactor to stable conditions. Additional SAMGs supplement the EOP, which in combination with the additional and diverse means for decay heat removal, make the CDF for SGTR low. The frequency of an ISLOCA initiating event was calculated to be 6.2×10^{-6}/ry. It was estimated that the probability of failing to prevent the ISLOCA sequence from proceeding to core damage was 0.01 (given that 6 hours is available to use the other unit HHSI), resulting in an ISLOCA CDF of 6.2×10^{-8}/ry. This improvement was based on taking credit for proceduralized operator actions and the shared HHSI system if available. The staff recognizes (NRC 1997b) that, in general, the contributions to total CDF from either SGTRs or ISLOCAs are relatively small for Westinghouse 3-loop PWRs. Further, the staff concludes, based on the points raised by FPL above, that the contributions from these initiators to core damage and risk for Turkey Point Units 3 and 4 are low, relative to other contributors.

FPL submitted an IPE of external events (IPEEE) by letter dated June 24, 1994 (FPL 1994). FPL did not identify any fundamental weaknesses or vulnerabilities to severe accident risk with regard to the external events related to seismic, fire, high winds, floods, transportation and nearby facility accidents, and other external hazards. In a Technical Evaluation Report, the NRC's contractor concluded that the IPEEE met the intent of Supplement 4 to Generic Letter 88-20 (ERI 1998). However, FPL used margins-type methodologies rather than PSA for addressing external events. Therefore, FPL chose to capture the potential risk benefits associated with external events by doubling the calculated benefits for a given SAMA. In the responses to the RAIs, FPL states that the CDF contribution from external events reported in the IPEEE submittal (tornado, transportation and nearby facilities and others) is estimated to be less than 7.0×10^{-7}/ry. FPL further argues that the PSA model used for the SAMA would make the risk contribution from these external events even lower due to a smaller seal LOCA probability (partially due to new seal O-rings for the RCPs) and the capability to cross-tie CCW from the opposite unit that was not credited in the Turkey Point IPEEE submittal. Even though the FPL approach in doubling of CDF to account for the calculated benefits for external events would provide a numerically reasonable estimate of the potential impact of external events, this approach fails to capture the benefits that could result from specific SAMAs that would be aimed at particular external events. Nevertheless, since the staff believes the search for external events vulnerabilities as part of the Turkey Point IPEEE did not identify any risk contributors that would benefit from potential SAMAs, the staff considers the present FPL approach to be adequate.

The Turkey Point Level 2 IPE submittal (FPL 1991) that was reviewed by NRC in 1992 has been modified recently to account for changes in the plant damage state frequencies, resulting from the Level 1 PSA modifications discussed earlier in this section, and changes brought about by additional research since the original Level 2 IPE was completed. This research includes the NRC studies on resolution of the direct containment heating (DCH) issue for Westinghouse PWRs (NRC 1996b), the steam explosions-induced containment failure issue (FPL 2001; NRC 1989), and other issues related to high pressure scenarios (i.e., induced SGTR and vessel thrust forces). The revision in the Level 2 PSA model as a result of the aforementioned changes, results in low probabilities of early containment failure modes and insignificant contributions to the overall risk. The staff concludes that the use of the FPL Level 2 model provides a sufficiently detailed characterization of containment response to support a license renewal SAMA analysis.

The process used by FPL to extend the containment performance (Level 2) portion of the PSA to an assessment of offsite consequences (Level 3 PSA) was reviewed. This included consideration of the source terms used to characterize fission product releases for each of 47 containment release modes and consideration of the major inputs and assumptions used in the offsite consequence analyses. FPL used the severe accident source terms presented in the Turkey Point Units 3 and 4 IPE for each of 47 containment release modes. The source terms were incorporated as input to the NRC-developed MACCS2 code. For radionuclides not reported in the IPE, fraction values were set to zero.

The release input parameters used in the Level 3 quantification as required for MACCS2 calculations were defined for Turkey Point. In general, it is assumed that the time (after accident initiation) when the accident reaches general emergency conditions, or when personnel can reliably predict that general emergency conditions will be attained, is about 4.9, 3, 2, and 10 hours, for late containment failure, early containment failure, ISLOCA, and SGTR scenarios, respectively. Early releases (including bypass sequences) are assumed to be more energetic as compared with other releases. All releases are assumed to be elevated (i.e., at a height of 30 m [99 ft]), and the assumed release time varies from about 4.9 hours (after scram) for early releases, to 24 hours for late releases. These assumptions are, for the most part, consistent with those of other studies, including NUREG-1150 (NRC 1990). Sensitivity calculations were also performed to assess the impact of releases due to inclusion of radionuclides not considered as part of the original IPE source term calculations (i.e., ruthenium, lanthanum, cerium, and barium). These sensitivity analyses (FPL 2001) showed an increase in the benefits (increase in risk-reduction potential) of about $3000 (from $801,500 to $804,500) when these radionuclides were added to the analysis with release fractions of 1.0×10^{-3} for key release modes. Thus, the impact is small.

The MACCS2 input used site-specific meteorological data processed from hourly measurements for one full year (1998). These data were collected at the site meteorological tower.

The staff (NRC 2001) requested information on the impact of the Turkey Point Units 3 and 4 power uprate and 18-month cycle burn-up on the radiological activity used in the risk analysis. In response, FPL (FPL 2001) stated that a comparison of the major core inventory reported in the MACCS2 end-of-cycle inventory for a 3412 MW(t) plant with the plant-specific estimates for the Turkey Point Units 3 and 4 power uprate conditions, shows an increase of less than 25 percent in the estimated baseline risk. On this basis, the staff concludes that this increase would need to be accounted for among the SAMA candidates that are not eliminated by qualitative screening.

The population distribution used as input to the MACCS2 analysis is based on 1990 census data. Population growth within a 80-km (50-mi) radius of the site was projected out to 2025 by using the computer program SECPOP90 (NRC 1997c). Projections were benchmarked with 1998 county-wide population estimates.

Evacuation modeling is based on a site-specific evacuation plan developed by FPL. It is assumed that the people within the evacuation zone (extending out to 16 km [10 mi] from the plant) would move at an average speed of approximately 12 m/s with a delayed start time of 5130 seconds. It is assumed that people beyond the 16-km (10-mi) radius would continue their normal work activities unless the 50 and 25 rem whole-body effective dose equivalent in 1 week limits are predicted to be exceeded. In these cases, relocation is assumed to occur after half a day and one day, respectively. A sensitivity analysis was performed that assumes that only 95 percent of the people within the evacuation zone would participate in the evacuation. The remaining 5 percent are assumed to go about their normal activities. This assumption is conservative relative to the NUREG-1150 study (NRC 1990), which assumes evacuation of 99.5 percent of the population within the emergency planning zone. It was further assumed in this sensitivity analysis that the evacuation speed was 1.0 m/s (3 ft/s) and that the evacuation delay time is 2 hours. The result is less than a 1-percent change in population dose and evacuation costs. Accordingly, the evacuation assumptions and analysis are deemed reasonable and acceptable for the purposes of SAMA evaluation.

Much of the site-specific economic data was provided by SECPOP90 (NRC 1997c) and used in the MACCS2 analyses. SECPOP90 contains a database extracted from U.S. Census Bureau (USCB) CD-ROMs (1990 census data), the 1992 Census of Agriculture CD-ROM Series 1B, the 1994 U.S. Census County and City Data Book CD-ROM, the 1993 and 1994 Statistical Abstract of the United States. These regional economic values were updated to 1997 using the Consumer Price Index and other data from the USCB and the Department of Agriculture.

Although some of the economic parameter values were based on values quoted in NUREG-1150 (NRC 1990), some were revised with more recent and/or site-specific data.

The SAMA analyses did not explicitly include the impact of uncertainties associated with severe accident risk at Turkey Point. In response to RAIs, FPL provided the results of the most recent PSA model for Turkey Point (NRC 2001) that demonstrate the uncertainties in the calculated CDF range from about 27 percent of the mean internal events CDF at the 5th percentile to about 2.5 times the mean internal events CDF at the 95th percentile (i.e., an order of magnitude spread in the calculated internal events CDF). The SAMA baseline CDF of 1.62 x 10^{-5}/ry corresponds to the 88th percentile of the latest CDF distribution (FPL 2001). In response to RAIs, FPL indicated that other factors that offset the higher CDF associated with higher failure rates, as reflected by the upper bounds of uncertainties, include modeling uncertainties and the cost estimates. In the response to an RAI on uncertainties, FPL argued that additional credit for severe accident management guidance "could have been taken to reduce the likelihood of containment failure and fission product release. Plant specific implementation of SAMA candidates may be complicated by space limitations, outage cost, regulatory requirements, seismic, fire and other considerations. These factors overestimate the benefit or underestimate the cost. It is concluded that the effect of considering these uncertainties associated with the SAMA cost-benefit estimate would, in effect, offset the uncertainties associated with the CDF estimates, thus making the conclusions robust. No SAMA candidates are considered cost-beneficial even when a higher-confidence CDF is used." (FPL 2001)

Consistent with NUREG/BR-0184, sensitivity studies performed using a 3-percent discount rate (versus the 7-percent rate used in the baseline analysis) show an increase in the benefits of potential SAMAs; however, this does not alter the ER conclusions on the unfavorable cost-benefit ratios for the considered severe accident management alternatives.

The staff concludes that the methodology used by FPL to estimate the CDF and offsite consequences for Turkey Point provides an acceptable basis from which to proceed with an assessment of the risk-reduction potential for candidate SAMAs. Further, the risk results that were calculated for Turkey Point are consistent with risk results for other nuclear power plants, when adjusted for differences in population, weather, and the magnitude and frequency of radiological releases. Accordingly, the staff bases its assessment of offsite risk on the CDF and offsite doses reported by FPL.

5.2.3 Potential Design Improvements

The process for identifying potential plant improvements, an evaluation of that process, and the improvements evaluated in detail by FPL are discussed in this section.

5.2.3.1 Process for Identifying Potential Design Improvements

FPL's process for identifying potential plant improvements consisted of the following three elements:

- a review of the Turkey Point IPE submittal and the updated PSA
- reviews of SAMA analyses submitted in support of original licensing and license renewal activities for other operating nuclear power plants and advanced light water reactor plants
- reviews of other NRC and industry documentation discussing potential plant improvements.

FPL's initial list of 167 candidate improvements was extracted from the process and is reported in Table F.2-1 in Appendix F of the ER (FPL 2000).

FPL performed a qualitative screening on the initial list of 167 SAMAs using the following criteria:

- The SAMA is not applicable to Turkey Point, either because the enhancement is only for boiling water reactors, the Westinghouse AP600 design, or pressurized water reactor ice condenser containments, or it is a plant-specific enhancement that does not apply at Turkey Point (Screening Criterion A), or
- The SAMA has already been implemented at Turkey Point (or the design meets the intent of the SAMA, as determined by plant review of each SAMA) (Screening Criterion B).

Based on the qualitative screening, 91 SAMAs were eliminated, leaving 76 subject to the final screening and evaluation process. Of the 91 SAMAs eliminated, 64 were eliminated because they had already been implemented at Turkey Point (or the design met the intent of the SAMA). The 76 remaining SAMAs are listed in Table F.2-2 of Appendix F of the ER (FPL 2000). The final screening process involves identifying and eliminating those SAMAs whose cost exceeded twice their benefit.

5.2.3.2 Staff Evaluation

FPL's efforts to identify potential SAMAs focused primarily on areas associated with internal initiating events. (This is reasonable, because external events only contribute a small amount to the total CDF.) The list of 76 SAMAs generally addressed the accident categories that are dominant CDF contributors (transients and small break LOCAs) or issues that tend to have a large impact on a number of accident sequences at Turkey Point Units 3 and 4. The preliminary review of FPL's SAMA identification process raised some concerns that plant-specific risk contributors were not adequately assessed. The staff requested (NRC 2001) additional plant-specific risk information (dominant minimal cutsets) to determine if any significant SAMAs might have been overlooked. Further, the staff requested specific information about the final SAMA candidates including the 16 SAMAs that were based on the Turkey Point plant-specific risk profile as modeled in the current PSA. Based on the initial submittal and the responses to the RAIs, it is the staff's opinion that FPL made a reasonable effort to search for potential SAMA candidates, using the knowledge and experience of its Probabilistic Risk Assessment (PRA) personnel; reviewing insights from the IPE, IPEEE, and other plant-specific studies; and reviewing plant improvements in previous SAMA analyses. The potential SAMA candidates included a balance of both hardware and procedural alternatives.

It is important to note that as a follow-up to IPE/IPEEE process, FPL has identified five potential enhancements to the plant's accident management capability, that were subsequently implemented, and were considered in more detail in the updated PSA as described below (FPL 2001):

- Replenishment of Refueling Water Storage Tank (RWST): This enhancement has been proceduralized in the Turkey Point EOP for loss of emergency coolant recirculation. In addition, the units can also share the high head safety injection systems, meeting the intent of RWST replenishment to prolong the injection for LOCAs by taking steps allowing use of the postulated nonaccident unit's RWST inventory.

- Primary System Depressurization: Procedures exist to use the sprays, auxiliary spray or pressure-operated relief values in the pressurizer to depressurize the Reactor Coolant System (RCS), under high pressure accident conditions. For beyond design basis severe accidents, a Severe Accident Guideline (SAG) has been developed that provides guidance for RCS depressurization to prevent high RCS pressure at a postulated vessel breach.

- AC Power Recovery: The importance of AC power recovery has been highlighted in the Turkey Point operator training. Hurricane procedures also emphasize the importance of verifying the performance of diesel generators (DGs). A more detailed time-dependent recovery analysis varying the mission time and the time to recover offsite power also allows more realistic quantification of the risk related to the loss of offsite power and SBO scenarios.

- Cross-connection of Component Cooling Water (CCW): This enhancement has been implemented at Turkey Point by providing specific steps in the applicable Off-Normal Operating Procedure to cross-connect the CCW between the two units. This action is also highlighted during operator training at Turkey Point.

- Manual Actuation of Containment Spray (Cavity Flooding): This enhancement has already been implemented at Turkey Point. A SAG has been developed in order to provide guidance for injecting water to the containment from a variety of sources including containment spray.

These enhancements were not included in the SAMA candidate identification process for Turkey Point Units 3 and 4 (FPL 2001) because they had been implemented at the facility.

The staff notes that the set of SAMAs submitted is not all inclusive, because additional, possibly even less expensive, design alternatives can always be postulated. However, the staff concludes that the benefits of any additional modifications are unlikely to exceed the benefits of the modifications evaluated and that the alternative improvements would not likely cost less than the least expensive alternatives evaluated, when the subsidiary costs associated with maintenance, procedures, and training are considered.

The staff concludes that FPL used a systematic and comprehensive process for identifying potential plant improvements for Turkey Point Units 3 and 4.

5.2.4 Risk-Reduction Potential of Design Improvements

FPL evaluated each of the 76 SAMAs remaining after the screening using a bounding technique. Each SAMA was assumed to completely eliminate all sequences that the specific enhancement was intended to address. Table 5-5 lists these bounding analyses, the respective assumptions, and the applicable SAMAs. FPL doubled the maximum benefit (based on the internal risk) to account for any unmodelled risk reduction that could also occur in external events, because there is no external events PSA model for Turkey Point Units 3 and 4. If the implementation costs were greater than two times the benefit, then the SAMA was screened from further consideration. The staff considers the use of a factor of two to implicitly account for the risk benefits associated with both internal and external events to be appropriate for the Turkey Point site.

The initial submittal (FPL 2000) did not give sufficient information regarding the actual risk reduction for the candidate SAMAs. For a given SAMA, all FPL provided was that the risk reduction was less than a given amount. Thus the staff could not determine how close the risk reduction was to the “less-than-value.” The staff requested more specific information in the RAIs (NRC 2001). FPL responded (FPL 2001) with a summary of the key risk-reduction attributes for each of the cases (see Table 5-5), including the total benefit that would be achieved from implementing the SAMA.

Table 5-5. SAMA Cost-Benefit Screening Analysis

Analysis Case and Description	SAMA	Total Benefit (Bounding)	Estimated Cost	Screening Conclusion
SAMAs Requiring Hardware Modifications that Exceed 2 x MAB				
Qualitative Assessment	33, 34, 35, 38, 39, 46, 53, 87, 115, 167	<$802K (MAB)	>2 x Benefit	Screened out
SAMAs Requiring Plant Hardware Modifications				
SEALCSF Eliminate all contribution from RCP seal LOCAs	7, 8, 9, 10, 11, 12, 13, 15, 16, 165	<$31K	>2 x Benefit	Screened out
No LOG (see Note 1) Eliminate all loss of grid events	47, 71, 75, 76 Comment: Industry estimates for 71 are $10M, and for 75 are $1M/mile	<$49K	>2 x Benefit	Screened out
SGCRVLP2 Eliminate all contribution from containment spray failure	31, 32, 48	<$177K	>2 x Benefit	Screened out
CI-OK Eliminate all contribution from early containment failure	88, 96, 157, 161	<$17K	>2 x Benefit	Screened out
SGFCSF Eliminate secondary decay heat removal failures	No SAMA identification numbers for this case			
HHDDPCSF Added two diesel-driven HHSI pumps (one for each unit)	117, 118, 124, 126,	<$131K	>2 x Benefit	Screened out

Table 5-5. (contd)

Analysis Case and Description	SAMA	Total Benefit (Bounding)	Estimated Cost	Screening Conclusion
NO-ISLOCA (see Note 2) Eliminate all contribution from ISLOCA	89, 90, 91, 92, 95, 96, 159, 160, 161 Comment: SAMAs 96 and 161 are also considered by base case CI-OK	<$17K	>2 x Benefit	Screened out
	SAMAs Requiring Procedure Modifications			
RABCSF Eliminate all contribution from failure of RAB ventilation	25	<$15.3K	>2 x Benefit	Screened out
NO-SGTR (Note 1) Eliminate all contribution from SGTR	79, 80, 81, 82, 83, 84, 85	<$1K	>2 x Benefit	Screened out
EDG5 Installation of another DG	57 Comment: Industry estimates installation of DG to be $431K - $25M	<$72K	>2 x Benefit	Screened out
OPERCSF Further increased operator training for critical human interactions	121	<$67K	>2 x Benefit	Screened out
OperCSI Provide capability to auto realign from injection mode to recirc mode	131	<$56K	~$450K	Screened out
	SAMAs Utilizing PRA (CDF or RRW (see Note 3)) as Argument for Elimination			
CDF <5E-07 or RRW = 1	59, 67, 97, 98, 99, 144, 148, 151, 156	~$0	>2 x Benefit	Screened out
RRW = 1.001, 1.005	135, 140	<$4.1K	>2 x Benefit	Screened out

Table 5-5. (contd)

Analysis Case and Description	SAMA	Total Benefit (Bounding)	Estimated Cost	Screening Conclusion
RRW = 1.008, 1.009	111, 123	<$8.1K	>2 x Benefit	Screened out
RRW = 1.016	134	<$13K	>2 x Benefit	Screened out
CDF contribution of <0.5%	152	<$4.1K	>2 x Benefit	Screened out
CDF contribution of <2%	129, 149	<$16.4K	>2 x Benefit	Screened out
CDF contribution of <2.5%	155	<$20.1K	>2 x Benefit	Screened out
CDF contribution of ~5%	146	<$41K	>2 x Benefit	Screened out
CDF contribution of 8.5%	101	<$68.2K	~$580K	Screened out

Note 1: Requires both plant hardware and procedure modifications.
Note 2: NO-ISLOCA SAMAs 89, 90, 91, 92, 95 require both plant hardware and procedure modifications
Note 3: RRW is the ratio of baseline risk to the risk calculated assuming complete elimination of the risk contribution addressed by the SAMA. Thus, a no-impact SAMA has a RRW of 1, and the relative impact of a SAMA is measured by RRW - 1.

The staff has reviewed FPL's bases for calculating the risk reduction for the various plant improvements and concludes that the rationale and assumptions for estimating risk reduction is reasonable and generally conservative (i.e., the estimated risk reduction is higher than what would actually be realized).

5.2.5 Cost Impacts of Candidate Design Improvements

FPL estimated the costs of implementing each SAMA through the application of engineering judgment, estimates from other licensees' submittals, and site-specific cost estimates. The cost estimates conservatively excluded the cost of replacement power during extended outages required to implement the modifications, and they did not include contingency costs associated with unforeseen implementation obstacles. Estimates based on modifications implemented or estimated in the past were presented in terms of dollar values at the time of implementation and were not adjusted to present-day dollars.

The minimum cost of making a procedural change (including training) was estimated at $30,000. The minimum hardware modification package was assumed to be $70,000. In response to the staff request for more specific cost information in the RAIs (NRC 2001), FPL (FPL 2001) provided a detailed cost breakdown for a digital feedwater control system (totaling $580,000) as an example.

The cost estimate minimums that are implied in Table F.2-2 of Appendix F of the ER (FPL 2000) were compared to estimates developed elsewhere for similar improvements, including estimates developed as part of other licensees' analyses of SAMAs for operating reactors and advanced light-water reactors. The FPL estimates were found to be consistent and reasonable for the SAMAs under consideration.

5.2.6 Cost-Benefit Comparison

The cost-benefit comparison as evaluated by FPL and the staff evaluation of the cost-benefit analysis are described in the following sections.

5.2.6.1 FPL Evaluation

The methodology used by FPL was based primarily on NRC's guidance for performing cost-benefit analysis, i.e., NUREG/BR-0184, *Regulatory Analysis Technical Evaluation Handbook*

(NRC 1997a). The guidance involves determining the net value for each SAMA according to the following formula:

$$\text{Net Value} = (\$APE + \$AOC + \$AOE + \$AOSC) - COE$$

where

$APE	=	present value of averted public exposure ($)
$AOC	=	present value of averted offsite property damage costs ($)
$AOE	=	present value of averted occupational exposure ($)
$AOSC	=	present value of averted onsite costs ($)
COE	=	cost of enhancement ($).

If the net value of a SAMA is negative, the cost of implementing the SAMA is larger than the benefit associated with the SAMA and it is not considered cost-beneficial. FPL's derivation of each of the associated costs is summarized below.

<u>Averted Public Exposure (APE) Costs</u>

The APE costs were calculated using the following formula:

APE = Annual reduction in public exposure (?person-rem/ry)
x monetary equivalent of unit dose ($2000 per person-rem)
x present value conversion factor (10.88, based on a 20-year period with a 7-percent discount rate).

As stated in NUREG/BR-0184 (NRC 1997a), it is important to note that the monetary value of the public health risk after discounting does not represent the expected reduction in public health risk due to a single accident. Rather, it is the present value of a stream of potential losses extending over the remaining lifetime (in this case, the renewal period) of the facility. Thus, it reflects the expected annual loss due to a single accident, the possibility that such an accident could occur at any time over the renewal period, and the effect of discounting these potential future losses to present value.

For the purposes of the initial screening, FPL calculated an APE of $234,207.

Averted Offsite Property Damage Costs (AOCs)

The AOCs were calculated using the following formula:

AOC = Annual CDF reduction
x offsite economic costs associated with a severe accident (on a per-event basis)
x present value conversion factor.

FPL cited an annual offsite economic risk of $22,850 based on the Level 3 risk analysis. This value, which corresponds to the frequency-weighted sum of the base offsite economic costs in Table F.1-5 of the ER (FPL 2000), appears to be higher than values for other sites and those presented in NUREG/BR-0184 (NRC 1997a). This higher value is primarily due to the relatively high population in the 80-km (50-mi) radius zone around the plant.

For the purposes of the initial screening, FPL calculated an AOC of $245,932.

Averted Occupational Exposure (AOE) Costs

The AOE costs were calculated using the following formula:

AOE = Annual CDF reduction
x occupational exposure per core damage event
x monetary equivalent of unit dose
x present value conversion factor.

FPL derived the values for averted occupational exposure based on information provided in Section 5.7.3 of NUREG/BR-0184 (NRC 1997a). Best estimate values provided for immediate occupational dose (3300 person-rem) and long-term occupational dose (20,000 person-rem over a 10-year cleanup period) were used. The present value of these doses was calculated using the equations provided in NUREG/BR-0184 in conjunction with a monetary equivalent of unit dose of $2000 per person-rem, a real discount rate of 7 percent, and a time period of 20 years to represent the license-renewal period.

For the purposes of the initial screening, FPL calculated an AOE of $6,153.

Averted Onsite Costs (AOSC)

The AOSCs include averted cleanup and decontamination costs, and averted power replacement costs. Repair and refurbishment costs are considered for recoverable accidents only and not for severe accidents. FPL derived the values for AOSC based on information provided in Section 5.7.6 of NUREG/BR-0184 (NRC 1997a).

Averted cleanup and decontamination costs (ACCs) are calculated using the following formula:

ACC = Annual CDF reduction
x present value of cleanup costs per core damage event
x present value conversion factor.

The total cost of cleanup and decontamination subsequent to a severe accident is estimated in NUREG/BR-0184 (NRC 1997b) as $\$1.5 \times 10^9$ (undiscounted). This value was converted to present costs over a 10-year cleanup period and integrated over the term of the proposed license extension.

For the purposes of the initial screening, FPL calculated an ACC of $188,082.

Averted power replacement costs PRCs are calculated using the following formula:

PRC = Annual CDF reduction
x present value of replacement power for a single event
x factor to account for remaining service years for which replacement power is required
x reactor power scaling factor.

For the purposes of the initial screening, FPL calculated an PRC of $127,818.

Thus, the total estimated present dollar value equivalent for severe accidents at Turkey Point Units 3 and 4 is about $802,000.

FPL Results

Of the 76 SAMAs considered, 75 were eliminated because the estimated costs were expected to exceed twice the total benefit. As discussed in Section 5.2.6.2 of this report, the remaining SAMA required additional analysis by the staff to demonstrate that the estimated costs would be expected to sufficiently exceed the estimated benefit so that it could be eliminated. The benefit was determined by assuming all risk for relevant internal events is eliminated. FPL doubled this value to bound additional benefits that might result for external events. The end result was that no SAMA candidates were found to be cost-beneficial.

FPL performed several sensitivity analyses to evaluate the impact of parameter choices on the analysis results. The sensitivity analyses included the calculation of candidate SAMA benefits using a 3-percent discount rate. There were no changes in the conclusions that resulted from the sensitivity assessments.

5.2.6.2 Staff Evaluation

The cost-benefit analysis performed by FPL was based primarily on NUREG/BR-0184 (NRC 1997a) and was executed appropriately. The staff believes that the candidates assessed have costs that are considerably higher than the associated benefits. One of the 76 candidates considered (a SAMA for hydrogen burn control) required additional analysis by the staff to demonstrate that the estimated costs would be expected to sufficiently exceed the estimated benefit so that it could be eliminated.

The staff specifically asked about the costs and benefits of using passive autocatalytic recombiners (PARs) for hydrogen control (NRC 2000). The motivation for this request was that PARs are being considered for hydrogen control at other nuclear power plants and that the FPL assessment for Turkey Point Units 3 and 4 indicates an opportunity to consider hydrogen burn mitigation because the conditional probability for containment failure from a hydrogen burn is large, about 25 percent given a core damage event. An effective system of PARs could reduce this percentage considerably. The potential risk-reduction benefit of PARs is estimated by the staff to be about $120,000, considering internal events only. (Including external events and assuming the factor of two used by FPL [FPL 2000] to account for external events, this benefit value increases to $240,000.) The value of $120,000 was derived by subtracting from the value calculated by FPL for the total risk benefit of about $800,000 (internal events) those contributions to the total risk benefit that would not be affected by the mitigation of core damage events, namely about $320,000 (e.g., onsite economic costs), thus yielding $480,000. About 25 percent of this value would be the benefit for the risk reduction from preventing containment failure from hydrogen burns, yielding $120,000. Doubling that to account for external events yields about $240,000 of benefit. (This benefit might be adjusted upward by up to 25 percent when considering the higher burnup of contemporary fuel cycles, and the resulting increased risk.) With the cost of a single PAR estimated at $45,000, a more detailed assessment may be warranted.

The staff considered both the response to the RAI and NRC analysis of similar issues in addressing the costs and the contribution of PARs to risk reduction. Based on the FPL response to an RAI, it appears that the contribution to late containment failure is overly conservative because actions associated with the SAMGs and the likelihood of a wet cavity were not credited in the FPL analysis. FPL stated that although “the estimated cost of the autocatalytic recombiner seems attractive, when additional requirements such as design, qualification, installation, testing, maintenance, procedures and training are included, the cost is expected to be substantially higher.” The staff agrees that the total cost, especially when considering multiple PAR units, would be substantially higher.

The staff has assessed and reviewed the role of hydrogen (and carbon monoxide) burns on late containment failures for other nuclear power plants with large-dry containments. Typically, hydrogen burns play a small role in late containment failure. As an example, the staff modeled the contribution of hydrogen combustion to containment failure for units with large dry containments as part of the NUREG-1150 study (NRC 1990). Table A.4-5 of NUREG/CR-4551 (NRC 1993) shows that the contribution of hydrogen combustion to late containment failure for the Zion plant is less than 0.1 percent.

When considering the realistic total costs of installation, training, and maintenance, it is the staff's opinion that the costs will be higher than the $120,000 to $240,000 range of the PAR benefit and considerably higher than the staff's estimate of PAR benefit for a typical PWR with a large, dry containment. Further the staff agrees with FPL that accounting for the wet cavity and accident mitigation actions in the SAMGs would reduce the probability of late containment failure and reduce the associated $120,000 to $240,000 range of risk-reduction benefits. This "accounting" would also bring Turkey Point Units 3 and 4 more in line with the results of other Level 2 PSAs for similar large-dry containment PWRs.

The staff concludes that PARs do not appear to be cost beneficial for Turkey Point Units 3 and 4. Therefore, it would not need to be implemented as part of license renewal pursuant to 10 CFR Part 54.

5.2.7 Conclusions

FPL compiled a list of 167 SAMA candidates using the SAMA analyses as submitted in support of licensing activities for other nuclear power plants, NRC and industry documents discussing potential plant improvements, and the plant-specific insights from the FPL IPE, IPEEE, and living PSA model. A qualitative screening removed SAMA candidates(1) that did not apply to Turkey Point Units 3 and 4 due to design differences, or (2) for which the SAMA had already been implemented at Turkey Point Units 3 and 4 (or the design meets the intent of the SAMA, as determined by plant review of each SAMA). A total of 64 SAMA candidates were eliminated because they had already been implemented at Turkey Point (or the design meets the intent of the SAMA, as determined by plant review of each SAMA) and 27 others were eliminated because they are not applicable to Turkey Point Units 3 and 4. Only 76 SAMA candidates remained after this screening process.

Using guidance in NUREG/BR-0184 (NRC 1997a), the FPL current PSA model and a Level 3 analysis developed specifically for SAMA evaluation, a maximum attainable benefit of about $802,000 was calculated. The PSA results used in the FPL SAMA analysis were calculated using internal event results only. Because Turkey Point Units 3 and 4 do not have an external events PSA model to account for the potential impact of external events on the results of the SAMA evaluations, FPL doubled the benefits for the purposes of comparison to the costs.

The staff reviewed the FPL analysis and concluded that the methods used and the implementation of those methods were sound.

Based on its review of the FPL SAMA analyses, the staff concurs that none of the candidate SAMAs are cost beneficial. This is based on conservative treatment of costs and benefits. This conclusion is consistent with the low residual level of risk indicated in the Turkey Point Units 3 and 4 PSA and the fact that Turkey Point has already implemented many plant improvements identified from the IPE and IPEEE process.

5.3 References

10 CFR 50. Code of Federal Regulations, Title 10, *Energy*, Part 50, "Domestic Licensing of Production and Utilization Facilities."

10 CFR 51. Code of Federal Regulations, Title 10, *Energy,* Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions."

10 CFR 54. Code of Federal Regulations, Title 10, *Energy,* Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."

10 CFR 100 Code of Federal Regulations, Title 10, *Energy*, Part 100, "Reactor Site Criteria."

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U.S. Nuclear Regulatory Commission (NRC). 2001. Letter from U.S. NRC to T. F. Plunkett, Florida Power & Light Company. Subject: "Request for Additional Information Related to the Staff's Review of Severe Accident Mitigation Alternatives for Turkey Point Units 3 and 4 (TAC Nos. MA9440 and MA9944) (January 31, 2001).