ERI/NRC 95-507

TECHNICAL EVALUATION REPORT ON THE "SUBMITTAL-ONLY" REVIEW OF THE INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS AT TURKEY POINT NUCLEAR PLANT, UNITS 3 AND 4

FINAL REPORT

Completed: January 1997
Final: January 1998

Energy Research, Inc.
P.O. Box 2034
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Work Performed Under the Auspices of the
United States Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, D.C. 20555
Contract No. 04-94-050

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EXECUTIVE SUMMARY

This technical evaluation report (TER) documents a "submittal-only" review of the individual plant examination of external events (IPEEE) conducted for the Turkey Point Nuclear Plant, Units 3 and 4. This technical evaluation review was performed by Energy Research, Inc. (ERI) on behalf of the U.S. Nuclear Regulatory Commission (NRC). The submittal-only review process consists of the following tasks:

- Examine and evaluate the licensee's IPEEE submittal and directly relevant available documentation.
- Develop requests for additional information (RAIs) to supplement or clarify the licensee's IPEEE submittal, as necessary.
- Examine and evaluate the licensee's responses to RAIs.
- Conduct a final assessment of the strengths and weaknesses of the IPEEE submittal, and develop review conclusions.

This TER documents ERI's qualitative assessment of the Turkey Point IPEEE submittal, particularly with respect to the objectives described in Generic Letter (GL) 88-20, Supplement No. 4, and the guidance presented in NUREG-1407.

Florida Power and Light Company (FPL) is the licensee of Turkey Point Unit 3 (Turkey Point-3) and Turkey Point Unit 4 (Turkey Point-4). The Turkey Point IPEEE submittal considers seismic; fire; and high winds, floods and other (HFO) external initiating events. The Turkey Point IPEEE was performed and reviewed by licensee and contractor personnel.

Licensee's IPEEE Process

With respect to the seismic IPEEE, Turkey Point Nuclear Plant is assigned to the reduced-scope seismic review category in NUREG-1407. FPL elected to implement a site-specific program for conducting the seismic IPEEE of Turkey Point Nuclear Plant. The site-specific program was developed primarily in response to GL 87-02 for resolution of Unresolved Safety Issue (USI) A-46 at Turkey Point, Units 3 and 4, and at St. Lucie Unit 1. The site-specific program represents a "scaled-back" approach to USI A-46 resolution. After meetings and correspondence with FPL, the NRC never designated its approval of the site-specific program for IPEEE resolution. Nonetheless, FPL proceeded with use of its site-specific program as the basis for conducting the seismic IPEEE. The site-specific seismic adequacy evaluation conducted for Turkey Point, Units 3 and 4, relied primarily on a plant walkdown that focused on component anchorage capability and the potential for adverse seismic-induced spatial interactions. A safe shutdown equipment list (SSEL) was developed based on a success path that assumes loss of offsite power (LOSP). The submittal does not describe the success path nor does it present a success path logic diagram. The evaluation approach does not explicitly address a small loss of coolant accident (LOCA). All components in the SSEL that had not been previously verified as having adequate seismic capacity were walked down by the seismic review team (SRT). The seismic review team used its judgment in assessing adequacy of seismic anchorage capacity and in identifying spatial interaction concerns. Components with obviously rugged anchorage were screened out; components with questionable seismic anchorage were identified as potential outliers. Spatial interaction concerns were also identified as potential outliers. The

potential outliers were analyzed in further detail, in order to make a final outlier designation. Resolutions were proposed for each designated outlier. Table 3.1 of this TER compares the features of FPL's site-specific IPEEE program against the elements of a reduced-scope evaluation that have been recommended in NUREG-1407. The table indicates that FPL's program has addressed only a subset of the recommended items/guidelines. The most significant differences in the two evaluation approaches are judged to be: a lesser scope of components in the FPL approach; a limited treatment of human actions in the Turkey Point study; and no treatment of containment systems in the FPL program. In addition, the format for documenting the seismic IPEEE did not follow the recommendations of NUREG-1407. It is important to note that – based on findings of a site audit, and pending follow-up action by the licensee – the NRC has reached closure on USI A-46 for Turkey Point. To a significant degree, the NRC's resolution of USI A-46 concerns has served as direct basis for formulating corresponding review findings in this TER for similar IPEEE concerns.

For the fire IPEEE, the licensee has addressed the issue of fire events at Turkey Point Units 3 and 4, as part of its individual plant examination (IPE) submittal for the plant. The licensee has conducted an extensive and detailed analysis for fire events at Turkey Point Nuclear Plant. Appendix R documentation has been used to establish fire-related plant features, including fire zones and areas. Only those items of safe shutdown equipment which have been defined in Appendix R are included in the fire analysis. To support the fire analysis, the licensee has conducted a fire walkdown. A consulting firm has assisted FPL analysts in the preparation of the fire analysis. The licensee has used the 1990 version of the Electric Power Research Institute (EPRI) fire-induced vulnerability evaluation (FIVE) methodology, and associated fire frequency and failure data, to evaluate the fire risk. Simple models have been used to evaluate fire damage and human recovery actions. For redundant train failure frequency evaluation, probabilistic risk assessment (PRA) models of the plant have been used.

For the HFO IPEEE, the licensee has adopted the methodology applicable for plants constructed prior to the issuance of the Standard Review Plan (SRP), with guidance provided in NUREG-1407. The analysis for hurricanes was included as part of the Turkey Point Nuclear Plant IPE submitted to the NRC in June 1991. A bounding analysis for external flooding was performed in conjunction with the hurricane analysis. The impact of tornado hazard was evaluated as the combined contributions of a high-wind component and a missile component. A family of mean wind-speed hazard curves was developed specifically for the Turkey Point site in NUREG-4762. Assessment of the missile component of tornado hazard relied upon evaluations in the Turkey Point PRA hurricane missile analysis. With respect to transportation and nearby facility hazards, FPL has determined that aircraft crashes, toxic chemical releases, and explosions cannot be eliminated from evaluation based on the SRP screening criteria, and hence, they were analyzed in more detail. Most hazards have been ultimately eliminated due to the distance of the plant site from the hazard source. The IPEEE analysis for aircraft crashes used the SRP methodology; results from the IPEEE analysis were compared with results from an analysis performed earlier by Sandia National Laboratories (SNL). Deterministic bounding analyses were used to screen out toxic chemical releases and explosions in the IPEEE. SNL had previously calculated the core damage contribution from lightning at Turkey Point Nuclear Plant. The results from that analysis did not meet the SRP screening criteria; however, the SNL analysis did not take into account the electrical system upgrade, nor the addition of two emergency diesel generators (for a total of four) at the site. FPL concluded that there are no unique vulnerabilities to lightning at Turkey Point Nuclear Plant.

Key IPEEE Findings

From the seismic IPEEE, the principal findings consist of qualitative walkdown insights, and few quantitative findings have been reported. The seismic adequacy evaluation for Turkey Point, Units 3 and 4, has revealed a significant number (35) of outliers for which safety enhancements have been proposed in response to USI A-46. In addition, the licensee is undertaking follow-up actions to resolve four other concerns identified by the NRC in its USI A-46 review process. Enhancements for IPEEE-only components (i.e., components outside the scope of USI A-46, but within the scope of IPEEE) were not addressed.

With respect to fire events, the licensee has reached the overall conclusion that there are no significant fire vulnerabilities at Turkey Point Nuclear Plant. With the exception of six areas, which include the control room, the cable spreading room and the intake cooling water structure, all fire zones and areas were screened out. The screening was based on either: (a) a lack of safe shutdown equipment in the given area; or (b) an estimated fire-induced core damage frequency (CDF) of less than 10^{-6} per reactor-year (ry) for the given area. The licensee cites several conservative assumptions that were made in the analysis, pertaining to assignments of fire occurrence rate and fire severity for the control room and cable spreading room, as well as the availability of alternate shutdown panels. The licensee concludes that the two most critical fire areas (i.e., control room and cable spreading room) do not pose a fire vulnerability. A qualitative analysis is presented for the reactor control rod equipment room, where the cables leading to motor-operated valves (MOVs) controlling the reactor coolant pump (RCP) seal flow are located. It is claimed in the IPEEE documentation that the operators will take control of component cooling water (CCW) and charging pump RCP seal injection, by manually opening (via hand-wheels) the appropriate valves. Fire propagation modeling has been performed for fire events initiated within the intake cooling water structure. In the event of a loss of the intake cooling water (ICW) system, RCP seal failure can be prevented by either: (1) re-activating the CCW system, by completing a cross-tie with the other plant unit; or (2) re-activating the "B" charging pump, by establishing a hose connection between the pump oil cooler and the service water system.

With respect to HFO events, FPL estimated core damage sequence frequencies, due to high winds (causing collapse of Unit 1 and 2 stacks), to be less than 10^{-7}/ry. The frequency of hurricane storm surge sufficient to inundate critical safety-related equipment was estimated to have an upper bound value of 10^{-4}/ry and a lower bound value of 10^{-6}/ry. Core damage sequences initiated by hurricane-generated missiles were evaluated to have frequencies below 10^{-7}/ry. The CDF result for external floods was found to be bounded by the result for hurricane surge. A CDF due to tornado wind and missiles was estimated to be 4.81×10^{-7} /ry for Unit 3, and 6.85×10^{-7}/ry for Unit 4. Air transportation accidents were screened out in the IPEEE on the basis of crash frequency. The most conservative value for crash frequency, associated with any given airport, was estimated to be 8.5×10^{-7}/yr. The estimated CDF as a result of a natural gas explosion in the intake structure was placed at 6.0×10^{-8}/ry, and hence, this event was screened out. The frequency of core damage resulting from a hydrogen explosion was estimated to be less than 10^{-7}/ry. All other sources of transportation or nearby facility hazards were screened out. In the IPEEE submittal, FPL presented the SNL estimate of CDF due to lightning. The resulting value was obtained as 2×10^{-6}/ry, but was calculated prior to an electrical system upgrade and the addition of two emergency diesel generators. FPL concluded that there are no unique vulnerabilities at Turkey Point Nuclear Plant related to lightning.

Generic Issues and Unresolved Safety Issues

As part of the seismic IPEEE, the licensee has considered the following other issues: Generic Issue (GI) 131, "Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants"; and USI A-45, "Shutdown Decay Heat Removal Requirements." With respect to GI-131, FPL has stated that lateral restraint was added to the movable support assembly of the flux mapping system in 1989. The licensee considers this issue to be closed. USI A-45 was not addressed directly in the licensee's seismic IPEEE submittal. The site-specific seismic adequacy evaluation study performed for Turkey Point, Units 3 and 4, considered a success path that depends on seismic capability of the auxiliary feedwater (AFW) system. The condensate storage tanks (CSTs) are the only components of the AFW systems that were actually included in the seismic evaluation; the submittal notes that the AFW pumps were previously reviewed for seismic adequacy as part of GL 81-14. In response to an RAI issued by the NRC as part of the USI A-46 review process, the licensee has indicated that there also exists a seismically qualified path for feed-and-bleed cooling at the plant.

As part of the fire IPEEE, the licensee has addressed both Sandia fire risk scoping study issues and USI A-45 concerns. For each of these issues, the licensee's evaluation has dealt with the significant concerns, and has not revealed any outstanding problem areas. However, the licensee's treatment of these issues did not address the possibility of equipment damage resulting from fire suppression system activation, nor did it develop equipment fragilities for fires.

As part of the HFO IPEEE, the licensee has addressed USI A-45. No information is provided in the IPEEE submittal, except for the reference to NUREG/CR-4762 (performed by SNL for Turkey Point), which was consulted by the licensee. Due to the absence of documentation of the walkdown process, it is unclear as to whether or not USI A-17 has been addressed by the licensee. Even though a direct discussion of GI-103 "Design for Probable Maximum Precipitation (PMP)" was not provided in the submittal, nevertheless, FPL noted that there are no concerns associated with the accumulation of water on roofs due to increased rainfall intensity.

Some information is also provided in the Turkey Point IPEEE submittal which pertains to generic safety issue (GSI)-147, GSI-148, GSI-156, and GSI-172.

Vulnerabilities and Plant Improvements

The licensee makes a general conclusion in the IPEEE submittal that there are no vulnerabilities to severe accident risk from external initiators. However, safety enhancements related to specific external initiators have been identified and proposed for resolution.

For seismic events, the plant-specific seismic adequacy evaluation for Turkey Point Nuclear Plant, Units 3 and 4, has revealed a number of noteworthy seismic findings, including a significant number (35) of components identified as seismic outliers. Relevant plant improvements or other resolution procedures have been proposed or implemented to address the identified outliers. Table 4.1 of this TER (reproduced from Table 5.0 of the Turkey Point Nuclear Plant seismic adequacy evaluation submittal) summarizes the outlier issues and the pertaining actions taken. The table indicates that 26 anchorage/support concerns, 12 interaction hazards, and 2 functional concerns were identified. In addition to these items, some cases of poor seismic "housekeeping" were observed during seismic walkdowns, and these were documented by the seismic review team. Also, as a result of an NRC site audit conducted during the USI A-46 review

process, the licensee has agreed to address additional issues related to: an anchorage concern, corrosion concerns requiring maintenance, a concern with interaction of station batteries, and the need for a strict housekeeping program. The licensee's submittal reports the results of high confidence of low probability of failure (HCLPF) calculations that were performed for a number of large storage tanks; these calculations initially produced the following results:

- Diesel Oil Storage Tank; HCLPF=0.21g (after upgrade)
- Condensate Storage Tanks; HCLPF=0.11g (after upgrade)
- Refueling Water Storage Tanks (RWSTs); HCLPF=0.11g (after upgrade)

Hence, even after implementation of safety upgrades, the reported HCLPF capacities for the CSTs and RWSTs were found to fall below the level of the design-basis earthquake for Turkey Point Nuclear Plant. In response to the NRC's USI A-46 review process, the licensee re-evaluated the capacities of CSTs and RWSTs, and reported them to meet the seismic design basis of 0.15g peak ground acceleration (PGA). In its review of the revised calculations, the NRC determined that the tank HCLPF capacities exceeded 0.12g PGA and were sufficiently close to the plant's seismic design level. The licensee has not reported HCLPF capacities for other outlier components being upgraded.

For fire events, although the CDF for control room fires is estimated to be 1.9×10^{-4}/ry, and there are several fire zones resulting in CDFs above 10^{-6}/ry, the licensee has concluded that there are no fire vulnerabilities at Turkey Point Nuclear Plant.

With respect to HFO events, FPL has stated that no vulnerabilities exist at the Turkey Point Nuclear Plant. The dominant HFO contributor to risk, which is identified in the IPEEE as storm surge, has been addressed by enhancing the "Natural Emergencies" Emergency Plant Implementation Procedure (EPIP) No. 20106. Within this procedure (which was not provided in the IPEEE submittal, and therefore, has not been reviewed), additional guidance has been provided to cope with the effects of severe storms. This procedure was in place and was cited by the licensee as contributing significantly to the preparation and mitigation of the effects of Hurricane Andrew. With respect to the effects of intense rainfall on roof loads, the submittal notes that the control building can withstand water accumulation up to 6 inches; above the level of 6 inches water spills over the sides of the roof. Increases in rainfall intensity would only result in a more rapid accumulation of water on the roof. The submittal also mentions that the roofs of other buildings at the plant would not accumulate water. In 1991, after discovery of the issue of storm surge, the existing flood walls and stop logs at Turkey Point were refurbished. In addition, the Unit 3 Emergency Diesel Generator (EDG) fuel oil transfer pump was raised to an elevation that reduces its vulnerability to storm surge. As a further consequence of the high winds analysis, and as a result of Hurricane Andrew, the Unit 1 and 2 (fossil plant) stacks were reinforced to a design wind load of 225 mph. Other modifications and procedural enhancements were implemented as a result of Hurricane Andrew, but these are not specified in the submittal.

Observations

In the seismic IPEEE, the site-specific program for seismic adequacy evaluations of Turkey Point, Units 3 and 4 addresses only a subset of the elements specified in NUREG-1407 as recommended items that should be considered in the seismic IPEEE of a reduced-scope plant. The evaluations do, nonetheless,

address some meaningful IPEEE-related concerns, and have resulted in a number of plant seismic safety enhancements. Given the NRC's resolution of related USI A-46 concerns, the following are considered to be the most significant remaining weaknesses of the seismic IPEEE submittal:

1. The SSEL is deficient,
2. A seismic containment performance assessment was not conducted,
3. The treatment of human actions is deficient,
4. The submittal does not provide adequate documentation of seismic-fire/flood interaction concerns, including component-specific walkdown findings,
5. The seismic IPEEE is incomplete with respect to reduced-scope evaluation recommendations found in NUREG-1407, and
6. The seismic IPEEE submittal is not documented in accordance with the format recommended in NUREG-1407, Appendix C.

In the fire IPEEE, the licensee has expended considerable effort in the preparation of the fire events analysis, and has presented the evaluation in summary form in its IPE submittal. The IPEEE report is not prepared in accordance to the guidelines provided in NUREG-1407, and it references the IPE report as the documentation source for the fire evaluation. The licensee has employed appropriate methodology and data bases for conducting the fire analysis. Based on the information presented, the following concerns could not be adequately verified:

1. Fire initiation frequency evaluation;
2. Assumptions used for fire propagation analysis;
3. Human actions for recovering from the effects of a fire;
4. Fire suppression system failure probability (i.e., if redundant cables and equipment are grouped together in a single compartment, the successful operation of the fire suppression system may not matter);
5. Probability of failure of the equipment unaffected by the fire; and
6. Cross-zone fire propagation, where active fire barriers are employed.

Certainly, notwithstanding these observations, the licensee has gained an important experience from the effort of inspecting the entire plant, with the exception of the containment, for potential fire vulnerabilities. As a final observation, the resolution of the Thermo-Lag issue may have a profound impact on the results of the IPEEE fire analysis.

With respect to HFO events, the licensee appears to have developed an appreciation of severe accident behavior for the Turkey Point Nuclear Plant. The licensee has gained an understanding of the overall

likelihood of core damage, under normal operating conditions, due to HFO events. The methodologies used to analyze each of the hazards associated with HFO events appear appropriate. As a result of the high wind analysis, site modifications and procedural enhancements were performed. Although the susceptibility of the site to high winds or storm surge was not cited as a vulnerability, FPL deemed it prudent to make related modifications. The most significant of these modifications consists of: the reinforcement of the Units 1 and 2 (fossil plant) stacks; the enhancement of the EPIP Procedure 20106, "Natural Emergencies"; and the refurbishment of the flood wall.

PREFACE

The Energy Research, Inc., team members responsible for the present IPEEE review documented herein, include:

<u>Seismic</u>

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Dr. John Lambright, of Lambright Technical Associates, contributed to the preparation of Section 2.4 following the completion of the draft version of this TER.

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ABBREVIATIONS

AC	Alternating Current
AFW	Auxiliary Feedwater
BOP	Balance of Plant
CCW	Component Cooling Water
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CST	Condensate Storage Tank
CVCS	Chemical Volume and Control System
EDG	Emergency Diesel Generator
EPIP	Emergency Plant Implementation Procedure
EPRI	Electric Power Research Institute
ERI	Energy Research, Inc.
FCIA	Fire Compartment Interaction Analysis
FIVE	Fire Induced Vulnerability Evaluation Method
FPL	Florida Power and Light Company
FRSS	Fire Risk Scoping Study
FSAR	Final Safety Analysis Report
GI	Generic Issue
GIP	Generic Implementation Procedure (SQUG)
GL	Generic Letter
GSI	Generic Safety Issue
HCLPF	High Confidence of Low Probability of Failure (Capacity)
HFO	High Winds, Floods and Other External Initiators
HHSI	High Head Safety Injection
HVAC	Heating, Ventilation and Air Conditioning
ICW	Intake Cooling Water
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
IRS	In-Structure Response Spectrum
LLNL	Lawrence Livermore National Laboratory
LOC	Level of Concern
LOCA	Loss of Coolant Accident
LOSP	Loss of Offsite Power
LPG	Liquid Propane Gas
MFW	Main Feedwater
MHE	Maximum Hypothetical Earthquake
MLW	Mean Low Water
MOV	Motor-Operated Valve
NRC	Nuclear Regulatory Commission
OL	Operating License
PC/M	Plant Change/Modification
PGA	Peak Ground Acceleration
PMP	Probable Maximum Precipitation
PORV	Power-Operated Relief Valve
PRA	Probabilistic Risk Assessment

PWO	Plant Work Order
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RCP	Reactor Coolant Pump
RLE	Review Level Earthquake
RWST	Refueling Water Storage Tank
SER	Staff Evaluation Report
SMA	Seismic Margin Assessment
SMM	Seismic Margin Methodology
SNL	Sandia National Laboratories
SQUG	Seismic Qualification Utility Group
SRP	Standard Review Plan
SRT	Seismic Review Team
SSE	Safe Shutdown Earthquake
SSEL	Safe Shutdown Equipment List
SSER	Supplemental Safety Evaluation Report
SSRAP	Senior Seismic Review and Advisory Panel
TER	Technical Evaluation Report
Turkey Point-3	Turkey Point Nuclear Plant, Unit 3
Turkey Point-4	Turkey Point Nuclear Plant, Unit 4
USI	Unresolved Safety Issue

1 INTRODUCTION

This technical evaluation report (TER) documents the results of the "submittal-only" review of the individual plant examination of external events (IPEEE) for the Turkey Point Nuclear Plant, Units 3 and 4 [1]. This technical evaluation review, conducted by Energy Research, Inc. (ERI), has considered various external initiators, including seismic events; fires; and high winds, floods, and other (HFO) external events.

The U.S. Nuclear Regulatory Commission (NRC) objective for this review is to determine the extent to which the IPEEE process used by the licensee, Florida Power and Light (FPL), meets the intent of Generic Letter (GL) 88-20, Supplement No. 4 [2]. Insights gained from the ERI review of the IPEEE submittal are intended to provide a reliable perspective that assists in making such a determination. This review involves a qualitative evaluation of the licensee's IPEEE submittal, development of requests for additional information (RAIs), evaluation of the licensee responses to these RAIs, and finalization of this TER.

The emphasis of this review is on describing the strengths and weaknesses of the IPEEE submittal, particularly in reference to the guidelines established in NUREG-1407 [3]. Numerical results are verified for reasonableness, not for accuracy; however, when encountered, numerical inconsistencies are reported. This TER complies with the requirements of NRC's contractor task order for an IPEEE submittal-only review.

The remainder of this section of the TER describes the plant configuration and presents an overview of the licensee's IPEEE process and insights, as well as the review process employed for evaluation of the seismic, fire, and HFO-events sections of the Turkey Point IPEEE. Sections 2.1 to 2.3 of this report present ERI's findings related to the seismic, fire, and HFO reviews, respectively. Sections 3.1 to 3.3 summarize ERI's conclusions and recommendations from the seismic, fire, and HFO reviews, respectively. Section 4 summarizes the IPEEE insights, improvements, and licensee commitments. Section 5 includes completed IPEEE data summary and entry sheets. Finally, Section 6 provides a list of references.

1.1 Plant Characterization

Turkey Point is a four-unit power generating facility; Units 1 and 2 burn fossil fuels, whereas Units 3 and 4 are powered by nuclear fuel. The plant is located on the shore of Biscayne Bay on the eastern (Atlantic) coast of Florida, near the southern tip of the peninsula (25 miles south of Miami). Each of the Turkey Point nuclear units is a three-loop Westinghouse pressurized water reactor (PWR), with a rated full-power core thermal output of 2,200 MWt and a net electrical output of 666 MWe. The containment for each unit is the large-dry type, and consists of a post-tensioned reinforced-concrete cylindrical shell and shallow dome, with a thick reinforced-concrete foundation slab; the entire reinforced-concrete structure is lined with steel plates for leak-tightness. Turkey Point Unit 3 went into commercial operation during December 1972, and Turkey Point Unit 4 began commercial operation during September 1973.

Turkey Point Units 3 and 4 share several common areas and systems. The control and auxiliary buildings, and the intake structure, are shared by both units. There is one control room and one cable spreading room for both units. These two rooms are situated at the upper elevations of the auxiliary building. Portions of the heating, ventilation, and air conditioning (HVAC) system are also shared among the two units, for the auxiliary and control buildings; however, additions and modifications to the plant have added a number of room-dedicated HVAC elements.

Shared systems between the two plants include: the auxiliary feedwater (AFW) system; the high head safety injection (HHSI) system; portions of the electric power system; the chemical and volume control system (CVCS); and (as noted above) portions of the HVAC system. All four units at Turkey Point share the same ultimate heat sink, a closed system of man-made cooling canals; however, each unit has its own intake cooling water (ICW) system. In addition, several systems can be cross-tied between the units, including: component cooling water (CCW), main feedwater (MFW), instrument air, and alternating current (AC) power (station blackout cross-tie). With respect to MFW, the fossil units (Units 1 and 2) can also be cross-tied with the nuclear units.

The IPEEE submittal report notes that a number of plant equipment items are not sheltered in buildings, and thus, are exposed to the environment. For example, there are no walls or roof enclosing the structure that supports the turbine and associated equipment.

Turkey Point Nuclear Plant was designed in the late 1960s. The maximum hypothetical earthquake (MHE) peak ground acceleration (PGA) is 0.15g horizontal, which defines the safe shutdown earthquake (SSE) for the plant. (The vertical component of the design ground motion is two-thirds of the horizontal component.) The SSE spectral shapes are the same for the two units; both units were designed for a Housner spectral shape. The plant is founded primarily on rock (permeable limestones).

For the IPEEE study, the following cutoff dates apply for establishing plant configuration and operating conditions: April 1990 was the freeze date for the seismic adequacy evaluation; mid-1990 was the freeze date for those external events included in the individual plant examination (IPE) analysis; and May 1993 was the freeze date for the remaining IPEEE elements.

1.2 Overview of the Licensee's IPEEE Process and Important Insights

1.2.1 Seismic

As documented in NUREG-1407, for seismic IPEEE purposes, Turkey Point is binned into the reduced-scope evaluation category. Rather than implementing a reduced-scope seismic evaluation, FPL has pursued the use of a site-specific program for conducting the seismic IPEEE of Turkey Point Nuclear Plant. This site-specific program was developed primarily for treatment of Unresolved Safety Issue (USI) A-46, and represents a "scaled-back" approach to achieving the objectives of GL 87-02 [4]. The justifications cited by FPL for performing a scaled-back analysis include: (a) very low probability of having an earthquake at the SSE level at FPL's plants; and (b) very low values of potential offsite releases and potential risk reductions given the postulated accident scenarios and seismic hazards.

FPL's scaled-back site-specific seismic adequacy program was approved, in concept, by the NRC for the purpose of addressing USI A-46. However, once FPL submitted the actual seismic adequacy evaluation studies [5], the NRC identified a number of concerns and potential deficiencies with the approach. The NRC's concerns are documented in its safety evaluation report (SER) pertaining to USI A-46 resolution [6]. A site investigation by the NRC was held at FPL's corporate headquarters and at the Turkey Point Nuclear Plant during the week of December 4-8, 1995 to help resolve the concerns noted in the NRC's SER. Many of the NRC concerns were alleviated by way of discussions with the licensee and its consultants; for other concerns, the licensee has agreed to implement corrective actions identified by the NRC. These items are documented in an NRC supplemental safety evaluation report (SSER) [7], wherein the NRC states that closure has been reached on all of the SER open items.

With respect to the seismic IPEEE, the NRC also had concerns, early on, with use of the FPL site-specific approach as a basis for resolving severe accident vulnerability issues. The NRC never gave its approval of FPL's program for treatment of the seismic IPEEE. Nonetheless, FPL proceeded with the use of the site-specific seismic adequacy evaluations for USI A-46 as the basis for conducting the seismic IPEEE.

Since the licensee's seismic IPEEE is essentially identical to its USI A-46 seismic adequacy evaluation study, and because many of the recommendations outlined in NUREG-1407 for a reduced-scope IPEEE are achieved if an acceptable USI A-46 evaluation has been performed, the NRC's SER and SSER determines (to a significant degree) that a corresponding review conclusion be made for similar IPEEE concerns. Hence, this TER indicates where a review finding has been based on NRC's safety evaluation for USI A-46.

FPL's approach to seismic evaluation relied primarily on plant walkdowns and on the use of seismic review team (SRT) judgment, supplemented with calculations, as needed, for resolving outliers. The walkdowns have addressed the following items: equipment seismic capacity versus demand, equipment construction adequacy, anchorage adequacy, seismic spatial interaction concerns, and seismic housekeeping concerns. The main overall elements of FPL's site-specific seismic adequacy evaluation have included:

- Project planning
- Selection of the seismic review team
- Preparatory work prior to seismic walkdowns
- Seismic capability walkdowns
- Limited seismic margin assessment (SMA) calculation work
- Resolution of outliers
- Peer review
- Documentation

FPL's approach to these aspects of the seismic IPEEE process for Turkey Point Nuclear Plant is discussed in Section 2.1.

FPL found no seismic vulnerabilities to potential severe accidents, but did report 35 outliers to be resolved. Additionally, in response to the NRC's USI A-46 review process, FPL agreed to resolve additional concerns related to: an anchorage problem, a maintenance issue pertaining to corrosion, a seismic interaction potential, and seismic housekeeping procedures.

1.2.2 Fire

Overall, the licensee has concluded that there are no significant fire vulnerabilities at Turkey Point Nuclear Plant, Units 3 and 4. With the exception of six areas, which include the control room, cable spreading room, and the intake cooling water structure, all fire zones and areas were screened out. This screening was based on either: (a) a lack of safe shutdown equipment in a given area; or (b) an estimated fire-induced core damage frequency of less than 10^{-6}/ry for a given area. The licensee cites several conservative assumptions that were made in the analysis, pertaining to assignments of fire occurrence rate and fire severity for the control room and cable spreading room, as well as the availability of alternate shutdown panels. The licensee concludes that the two most fire risk significant areas (i.e., control room and cable spreading room) do not pose a fire vulnerability. A qualitative analysis is presented for fire assessment

of the reactor control rod equipment room, where the cables leading to motor-operated valves (MOVs) which control the reactor coolant pump (RCP) seal flow are located. It is claimed in the IPEEE documentation that the operators will take control of component cooling water (CCW) and charging pump RCP seal injection, by manually opening (via hand-wheels) the appropriate valves. Fire propagation modeling has been performed for fire events initiated within the intake cooling water structure. In the event of a loss of the intake cooling water (ICW) system, RCP seal failure can be prevented by either: (1) re-activating the CCW system, by completing a cross-tie with the other plant unit; or (2) re-activating the "B" charging pump, by establishing a hose connection between the pump oil cooler and the service water system.

The licensee has addressed Sandia fire risk scoping study issues and USI A-45 concerns. For each of these issues, the licensee's evaluation has dealt with the significant concerns, and has not revealed any outstanding problem areas. However, the licensee's treatment of these issues does not discuss the possibility of equipment damage resulting from spurious/inadvertent fire suppression system activation, nor does it develop equipment fragilities for fire conditions.

1.2.3 HFO Events

The IPEEE submittal has employed a comprehensive list of potential external hazards to assist in identifying external events for which more detailed analyses are judged to be needed. These events were found to include: high winds and tornadoes, external flooding, transportation and nearby facilities accidents, and lightning. These external events have been analyzed by a progressive screening approach to which quantitative methods have been applied, as necessary. In several instances, the plant probabilistic risk assessment (PRA) model was re-assessed in order to thoroughly evaluate the impacts of the external hazard. No vulnerabilities were cited by FPL as a result of the HFO IPEEE effort.

1.3 Overview of Review Process and Activities

In its qualitative review of the Turkey Point IPEEE, ERI focused on the study's completeness in reference to NUREG-1407 guidance; its ability to achieve the intent and objectives of GL 88-20, Supplement No. 4; its strengths and weaknesses with respect to the state-of-the-art; and the robustness of its conclusions. This review did not emphasize confirmation of numerical accuracy of submittal results; however, any numerical errors that were obvious to the reviewers are noted in the review findings. The review process included the following major activities:

- Completely examine the IPEEE and related documents
- Develop a preliminary TER and RAIs
- Examine responses to the RAIs
- Finalize this TER and its findings

Because these activities were performed in the context of a submittal-only review, ERI did not perform a site visit or an audit of either plant configuration or detailed supporting IPEEE analyses and data. Consequently, it is important to note that the ERI review team did not verify whether or not the data presented in the IPEEE matches the actual conditions at the plant, and whether or not the programs or procedures described by the licensee have indeed been implemented at Turkey Point.

1.3.1 Seismic

In conducting the seismic review, ERI generally followed the emphasis and guidelines described in the report, *Individual Plant Examination of External Events: Review Guidance* [8], for review of a seismic margin assessment, and the guidance provided in the NRC report, *IPEEE Step 1 Review Guidance Document* [9]. In addition, on the basis of the Turkey Point IPEEE submittal, ERI completed data entry tables developed in the Lawrence Livermore National Laboratory (LLNL) document entitled *"IPEEE Database Data Entry Sheet Package"* [10].

In its Turkey Point IPEEE seismic review, ERI examined the following documents:

- The licensee's IPEEE submittal [1], Sections 1, 2, 3, 4, 6, 7, and 8
- The USI A-46 seismic adequacy evaluation of Turkey Point, Units 3 and 4 [5]
- Section 3.4.7 of the Turkey Point PRA [11]
- The NRC's safety evaluation report [12] of the Turkey Point IPE
- The NRC's SER [6] and supplemental SER (SSER) [7] of the USI A-46 submittals for Turkey Point, Units 3 and 4, and St. Lucie Unit 1
- Section 3.9.3 of the IPE submittal for Turkey Point, Units 3 and 4 [11]
- The licensee's response [13] to the RAIs generated as part of the initial submittal review

The IPEEE submittal [1] itself contains only one page of discussion related to seismic evaluation. Consideration of the seismic adequacy evaluation study (Reference [5]) and the NRC's evaluation [6,7] of the licensee's USI A-46 submittal constituted the most significant element of the present seismic review. The checklist of items identified in Reference [8] was generally consulted in conducting the seismic review. Some of the primary considerations in the seismic review have included (among others) the following items:

- Were appropriate walkdown procedures implemented, and was the walkdown effort sufficient to accomplish the objectives of the seismic IPEEE?

- Was the development of success paths performed in a manner consistent to prescribed practices? Were random and human failures properly considered in such development?

- Were component demands assessed in an appropriate manner, using valid seismic motion input and structural response modeling, as applicable? Was screening appropriately conducted?

- Were capacity calculations performed for a meaningful set of components, and are the capacity results reasonable?

- Does the submittal's discussion of qualitative assessments (e.g., containment performance analysis, seismic-fire evaluation) reflect reasonable engineering judgment, and have all relevant concerns been addressed?

- Has the seismic IPEEE produced meaningful findings, has the licensee proposed valid plant improvements, and have all seismic risk outliers been addressed?

It is important to note that, in a number of instances, IPEEE review findings have been reported on the basis of consistency with related findings in NRC's SER [6] and SSER [7] for USI A-46, rather than on the basis of a separate review.

1.3.2 Fire

During this technical evaluation, ERI reviewed the fire-events portion of the IPEEE for completeness and consistency with past experience. This review was based on consideration of Sections 1, 2, 4, 6, 7, 8 and 9 of Reference [1], Sections 3.7 and 3.9 of Reference [11], and on the licensee responses to fire-related RAIs [13]. In addition, a set of layout drawings [14] pertaining to fire protection were available for review. The guidance provided in References [8,9] was used to formulate the review process and organization of this document. The data entry sheets used in Section 5 have been completed in accordance with Reference [10].

The process implemented for ERI's review of the fire IPEEE included an examination of the licensee's methodology, data, and results. ERI reviewed the methodology for consistency with currently accepted and state-of-the-art methods. The data element of a fire IPEEE includes, among others, such items as:

- Cable routing
- Fire zone/area partitioning
- Fire occurrence frequencies
- Event sequences
- Fire detection and suppression capabilities

For a few fire zones/areas that were deemed important, ERI also verified the logical development of the screening justifications/arguments (especially in the case of fire-zone screening) and the computations for fire occurrence frequencies and core damage frequencies (CDFs). Rather than perform a completely independent set of calculations, however, the review team used its experience and comparisons of other plants and fire evaluation results, in order to judge the accuracy and completeness of the information provided by the licensee. Special attention was directed to: (1) the screening methodology, because a trend to prematurely screen out potentially significant areas or to inadequately justify screening out an area, has emerged as a common problem among past fire PRAs and IPEEE analyses; and (2) the licensee's assumptions, because the results of many studies are unduly influenced by assumptions made to simplify or introduce conservatisms.

1.3.3 HFO Events

The review process for HFO events closely followed the guidance provided in the report entitled *IPEEE Step 1 Review Guidance Document* [9]. This process involved examinations of the methodology, the data used, and the results and conclusions derived in the submittal. Sections 1, 2, 5, 6, 7, 8 and 9 of the IPEEE submittal [1], and licensee responses to RAIs [13], were examined in this HFO-events review. The IPEEE methodology was reviewed for consistency with currently accepted practices and NRC recommended procedures. Special attention was focused on evaluating the adequacy of data used to estimate the frequency of HFO events, and on confirming that any analysis of Standard Review Plan (SRP) conformance was appropriately executed. In addition, the validity of the licensee's conclusions, in consideration of the results reported in the IPEEE submittal, was assessed. Also, bounding-analysis and PRA results pertaining to frequencies of occurrence of hazards and estimates of conditional probabilities

of failure, were checked for reasonableness. Review team experience was relied upon to assess the validity of the licensee's evaluation.

2 CONTRACTOR REVIEW FINDINGS

2.1 Seismic

A summary of the licensee's seismic IPEEE process has been described in Section 1.2. Here, the licensee's seismic evaluation is examined in detail, and discussion is provided regarding significant observations encountered in the present review.

2.1.1 Overview and Relevance of the Seismic IPEEE Process

a. Seismic Review Category and Review Level Earthquake (RLE)

Turkey Point Nuclear Plant is located in an area of low seismicity, on the shore of Biscayne Bay (eastern coastal Florida) near the southern tip of the peninsula (25 miles south of Miami). Turkey Point Nuclear Plant was designed in the late 1960s; both units are in the USI A-46 program.

The maximum hypothetical earthquake PGA for Turkey Point Nuclear Plant is 0.15g horizontal, which defines the SSE for the plant. (The vertical component of the design ground motion is two-thirds of the horizontal component.) The SSE spectral shapes are the same for the two units; both units were designed for a Housner spectral shape. The plant is founded primarily on rock (permeable limestones).

Due to the low seismic hazard at the site, Turkey Point has been designated as a reduced-scope plant in NUREG-1407. The RLE is equivalent to the SSE.

b. Seismic IPEEE Process

The licensee has implemented a site-specific seismic adequacy evaluation program based on a methodology it has compiled for executing its USI A-46 resolution program at Turkey Point Units 3 and 4, and at St. Lucie Unit 1. (The NRC has determined, pending appropriate follow-up action by the licensee, that USI A-46 has been adequately resolved for Turkey Point [6,7].) The licensee claims that its USI A-46 seismic evaluation process conforms with the Optional Methodology of Paragraph 3.3 in NUREG-1407. However, the program was never actually approved by the NRC.

c. Review Findings

The IPEEE process is not fully consistent with the recommended guidelines of NUREG-1407 for Turkey Point. FPL's seismic programs for Turkey Point Units 3 and 4 address only a portion of the seismic IPEEE elements/concerns for a reduced-scope plant. The Turkey Point IPEEE submittal is essentially identical to the USI A-46 submittal. Hence, the concerns/findings documented by the NRC in its review for USI A-46 are applicable to a number of aspects of the seismic IPEEE.

Nonetheless, the fact that FPL's seismic adequacy evaluation program departs from a complete reduced-scope assessment is viewed to be a significant weakness. The overall seismic IPEEE methodology employed by FPL has only a limited potential to achieve IPEEE objectives, and to assess severe accident vulnerabilities at Turkey Point Nuclear Plant.

2.1.2 Success Paths and Component List

Success was defined, for purposes of identifying a success path, as the ability to achieve and maintain a hot shutdown condition for 8 hours. Loss of offsite power was assumed in choosing the success path. In addition, a design basis earthquake was assumed not to trip the reactor.

The primary elements of the chosen success path include: supervisory and control function requirements, requirements of decay heat removal via the AFW system, emergency electrical power requirements, chemical and volume control requirements, and equipment cooling (ultimate heat sink) requirements via the CCW and ICW systems.

The submittal states that all active equipment pertaining to the success path were identified in developing a safe shutdown equipment list (SSEL). Some passive components, such as tanks and heat exchangers, were also included in the SSEL. A significant number of components (e.g., AFW pumps) were removed from the SSEL because they had been previously reviewed for seismic adequacy in another program. (Similarly, potential interaction concerns that involved block walls were considered resolved if the walls were previously addressed under IE 80-11 [15]). The resulting SSEL defines the set of components considered in plant walkdowns.

FPL's seismic adequacy evaluation does not clearly identify the chosen success path, nor does it present a success path logic diagram. Only one success path was involved in developing the SSEL, and only a limited set of components were identified for each major success-path function. The study did not explicitly address a small-break loss of coolant accident (LOCA) in the development of the success path and SSEL. The SSEL considers active components and a partial list of passive components. The success criterion used in the FPL study is the ability to achieve and maintain hot shutdown for a time period of only 8 hours, rather than the recommended 72 hours. However, in response to RAIs raised by the NRC in its USI A-46 review process, FPL indicated that the plant has multiple (albeit non-seismically qualified) water sources that could provide cooling for 72 hours. In addition, FPL indicated that the plant has the (seismically qualified) capability of indefinitely long feed-and-bleed cooling.

Thus, the equipment list developed in the FPL study appears to be very limited, and considers only a subset of components that should be evaluated in a reduced-scope assessment.

2.1.3 Non-Seismic Failures and Human Actions

a. *Overall Approach*

The Turkey Point Nuclear Plant seismic adequacy study notes that a review of operating procedures was performed to verify the equipment list and to identify any equipment which might be required to bring the reactor from 100% power to hot shutdown. Additionally, operating procedures to shut down the reactor, take the reactor to hot shutdown, to respond to reactor trip, and to respond to loss of offsite power were reviewed. No mention is made of specific non-seismic failures or human actions that might limit the capability of the chosen success path.

b. Screening Criteria

Random and operator failure rates were not reported; no screening criteria were applied with respect to non-seismic failures and human actions.

c. Review Findings

According to NUREG-1407, candidate success paths should be screened to insure that impacts of non-seismic failures and human actions will not be controlling factors inhibiting the likelihood of successful hot shutdown. FPL's seismic evaluation has not identified the specific random failures and human actions which might compromise the integrity of the chosen success path. Hence, the licensee's study is inadequate in its treatment of non-seismic failures and human actions, which is thus viewed to be a weakness of the study.

2.1.4 Seismic Input

Seismic inputs for evaluation studies of Turkey Point, Units 3 and 4, were defined by an SSE (i.e., MHE) spectrum and other plant-specific design-basis commitments in the final safety analysis report (FSAR). For both Turkey Point Units 3 and 4, the SSE is identified by a Housner spectral shape anchored to a PGA level of 0.15g (design horizontal component). The design vertical component of motion is taken to be two-thirds of the design horizontal component. The SSE/FSAR requirements are applicable for rock-site conditions (the plant is founded primarily on permeable limestones).

NUREG-1407 indicates that the SSE ground response spectra should be used to define input to structures, and for computing in-structure response spectra. FPL's seismic adequacy evaluation program uses the SSE spectrum or FSAR in-structure spectra as the basis for defining seismic input for components. Hence, the licensee's definition and use of seismic input is consistent with the guidelines of NUREG-1407 for a reduced-scope plant.

2.1.5 Structural Responses and Component Demands

Turkey Point Nuclear Plant had some existing floor response spectra curves, which were used to define demands for a number of the SSEL components. For components where existing floor response spectra were not available for assessing demands, estimates of component demands were made based directly on the SSE (MHE) spectrum. The approach for assessing such demands (for equipment less than 40 feet above grade) was to: (a) take the peak spectral acceleration from the 5%-damped SSE spectrum, (b) multiply this peak value by 1.5 to account for building amplification, and (c) multiply again by a factor of 1.25 for conservatism.

NUREG-1407 indicates that existing FSAR in-structure spectra, based on SSE input and FSAR licensing criteria, may be used for evaluating component demands. In the FPL seismic adequacy studies, FSAR in-structure spectra were used, when available, to establish equipment demands. When in-structure spectra were not available, a generally conservative procedure based on scaling the peak SSE spectral acceleration was used to define component demands. The licensee's development of component demands thus appears consistent, to a significant degree, with the guidelines of NUREG-1407 for a reduced-scope plant. Additionally, the NRC has accepted this aspect of the licensee's analysis for USI A-46 resolution [7].

2.1.6 Screening Criteria

Screening for the Turkey Point seismic evaluation study has not followed the formal procedures described in the generic implementation procedure (GIP) [16] or in Electric Power Research Institute (EPRI) NP-6041 [17], as recommended in NUREG-1407. Rather, the procedures described in Reference [18], the Senior Seismic Review and Advisory Panel (SSRAP) document, have generally been implemented.

Whether GIP, EPRI NP-6041, or other procedures are used for screening, screening caveats must be observed, anchorage capacity checks must be performed, and spatial interaction issues must be appropriately assessed. Additionally, in any screening procedure, SRT judgment plays the major role in component evaluations.

FPL's screening approach has been based primarily on SRT judgment, on comparisons of estimated anchorage capacity versus SSE-consistent demand, and on insights derived by the SSRAP. Although the licensee's approach to screening does not conform precisely to the recommendations of NUREG-1407 for a reduced-scope plant, it is judged to be a reasonable process that substantially achieves the significant intent of component screening.

2.1.7 Plant Walkdown Process

a. Preparatory Work

A pre-walkdown of the plant was performed to help the seismic review team (SRT) members identify what information and assistance would be needed during the seismic capability walkdown. FPL engineers gathered generic and equipment-specific documentation as deemed necessary by the SRT. In addition, FPL staff familiar with plant systems developed the list of equipment to be walked down.

b. Seismic Capability Walkdown

Plant walkdowns were conducted by an SRT consisting of three highly experienced walkdown experts. The seismic adequacy evaluation studies have relied heavily on the judgment of these engineers. During the walkdown, FPL provided staff engineers to help support the SRT members, primarily in obtaining additional plant information that was needed on a case-by-case basis. The actual duration of seismic walkdowns is not mentioned in the documentation.

Four considerations were addressed in the plant walkdown screening effort: (1) equipment seismic capacity versus demand, (2) construction adequacy of equipment, (3) anchorage adequacy, and (4) seismic spatial interaction concerns. The walkdown also made note of concerns related to: (5) general seismic "housekeeping" issues. Each of these aspects of plant walkdowns and component screening is described briefly below.

1. Equipment seismic capacity versus demand - This screening item pertains to identification of seismic adequacy problems that could be inherent to specific types of unqualified seismic equipment. These encompass the types of problems that would be found in a qualification test, including: functional problems, internally fragile elements, and inadequate overall structural resistance of a cabinet.

The Turkey Point seismic adequacy evaluation treated this item in a generic way based on findings of the SSRAP, as documented in Reference [18]. It was demonstrated in the evaluation studies that (for use with respect to equipment having a natural frequency greater than 8 Hz and located less than 40 feet above grade) the SSRAP bounding spectrum envelopes plant SSE spectra over the entire frequency range. It was also demonstrated that (for use with respect to equipment having a natural frequency less than 8 Hz or located more than 40 feet above grade) the SSRAP bounding spectrum multiplied by 1.5 enveloped plant floor response spectra. Since the bounding spectrum represents an experience-based seismic ruggedness threshold for unqualified nuclear power plant equipment, the FPL study concludes that seismic capacity versus demand was judged acceptable for all plant components. The plant walkdowns, therefore, did not give much attention to this screening item, on a component-by-component basis.

2. *Construction adequacy of equipment* - This screening item pertains to identification of seismic adequacy problems that could be attributed to the configuration or manner of construction/installation of the equipment at the plant. Generally speaking, the as-built configuration of equipment can be considered adequate, provided that certain caveats have been considered and satisfied.

   FPL reasoned that, due to low seismicity at FPL plant sites, specific caveats did not need to be addressed for each type of equipment. The seismic evaluation study further noted that SRT members are experts in the area of seismic adequacy of equipment, and that they noted any equipment-specific details that they felt were seismically vulnerable.

3. *Anchorage adequacy* - This screening item pertains to identification of seismic adequacy problems that are due to non-existent or weak anchorage. The constructed anchorage configuration can be considered as a caveat to be addressed in the evaluation of all components. It is a special caveat, however, because its treatment usually requires more than just a visual inspection; the expected demand on the anchorage and a numerical estimate of anchorage capacity are often needed to satisfy anchorage caveats.

   In the seismic adequacy evaluations, SRT judgment was used to screen out "obviously rugged" anchorages. Otherwise, a numerical estimate of seismic adequacy of anchorage components was obtained and compared against component anchorage demand. Any problems noted with anchorage capacity were designated as potential outliers to be resolved.

4. *Seismic spatial interaction concerns* - This screening item pertains to the identification of physical effects that could independently compromise the performance of an otherwise well-installed seismically adequate component. Such physical effects include: objects impacting equipment in any manner, conduit pull-out due to inadequate flexibility of lines attached to equipment, block wall collapses, etc.

   During the walkdowns, SRT members looked for, and made note of (on walkdown work sheets), any potential seismic spatial interaction concerns; identified concerns were designated as potential outliers.

5. *Seismic housekeeping concerns* - This walkdown item pertains to situations that, although not leading to failure of an important safety-related component, can exacerbate problems and/or inhibit operator effectiveness following an earthquake.

   Any instances of poor seismic housekeeping observed by SRT members were noted and reported to FPL.

Among these five walkdown items, primary consideration was given to assessing anchorage adequacy and to identifying seismic spatial interactions.

c. Review Findings

NUREG-1407 recommends the use of GIP or EPRI NP-6041 walkdown procedures. The Turkey Point walkdown has implemented procedures substantially similar to these, perhaps allowing for somewhat greater latitude in the use of expert judgment. Due in large part to the exceptional qualifications of the SRT members, and the NRC's acceptance of the seismic walkdown for USI A-46 resolution, the licensee's walkdown process is considered to be adequate in identifying outliers among those components that have been included in the scope of walkdowns.

2.1.8 Evaluation of Outliers

a. Overall Approach

The seismic adequacy evaluations do not make a clear distinction between "outlier" and "potential outlier." All items not screened out by the SRT were addressed in some manner by FPL. For potential anchorage outliers (i.e., those anchorage concerns screened in by the SRT during plant walkdowns), more-detailed calculations were performed to better determine seismic adequacy. Any component having inadequate/low anchorage capacity was identified as an outlier requiring resolution by FPL.

b. High Confidence of Low Probability of Failure (HCLPF) Calculations

For Turkey Point Units 3 and 4, HCLPF calculations were performed for many large, flat-bottom tanks. No HCLPF calculations were performed for block walls identified to be a potential interaction problem. (The seismic adequacy evaluations rely on earlier IE 80-11 calculations.)

c. Review Findings

For some components that were screened-in at Turkey Point Nuclear Plant, capacity calculations were performed to demonstrate whether or not the component met the FSAR demand (or the conservative approximation to the FSAR demand). For components identified as final outliers, however, the outlier assessment was often readily made (without calculation) due to an obviously deficient condition (e.g., no anchorage, poor anchorage, poor bracing, etc.). For each final outlier noted, FPL proposed a corrective measure and generated a plant change/modification (PC/M) package or a plant work order (PWO).

The licensee's walkdown process is judged to be adequate in identifying outliers among those (limited set of) components that have been included in the scope of walkdowns.

2.1.9 Relay Chatter Evaluation

NUREG-1407 indicates that completion of the USI A-46 review requirements for relay chatter evaluation will satisfy the IPEEE intent for reduced-scope plants that are also USI A-46 plants. For reduced-scope plants that are not also USI A-46 plants, no relay chatter evaluation is necessary.

The licensee's IPEEE submittal does not mention a relay chatter evaluation for Turkey Point. However, during NRC's USI A-46 review, it was revealed that FPL had assessed bad actor relays, verified mountings of relays, and demonstrated that there were no deleterious effects of chatter of bad actor relays. The NRC accepted the licensee's relay evaluation for USI A-46 resolution, and hence, the NUREG-1407 recommendation for the seismic IPEEE is satisfied.

2.1.10 Soil Failure Analysis

NUREG-1407 states that no evaluation of soil failures is required for a reduced-scope plant. Correspondingly, the licensee has not performed such an analysis.

2.1.11 Containment Performance Analysis

For reduced-scope plants, NUREG-1407 requests that performance of containment and containment systems should be addressed. Components necessary to achieve successful accident mitigation need to be included in the scope of seismic walkdowns and outlier evaluation.

FPL did not include the containment structures or containment systems in its seismic adequacy evaluation of Turkey Point, Units 3 and 4. Hence, the licensee's seismic adequacy study of Turkey Point Nuclear Plant is not responsive to the NUREG-1407 request for a containment performance analysis.

2.1.12 Seismic-Fire Interaction and Seismically Induced Flood Evaluations

a. Evaluation of Seismic-Fire Interactions

Section 3.7.4 of the Turkey Point PRA (IPE submittal report) discusses seismic-fire interactions. The topic of seismic-fire interactions is one element of the Sandia fire risk scoping study (FRSS) issues. The IPEEE submittal states that all Sandia FRSS issues are more than adequately covered through the Turkey Point Fire Protection Program. In terms of details of the seismic-fire evaluation, however, the submittal indicates only that: "Essentially, the II/I criteria was applied to fire systems whose failure could affect operation of safety-related systems." Section 2.2.12 of this TER indicates that the licensee has addressed issues pertaining to seismically induced fires and inadvertent actuation of fire suppression systems in the fire analysis and treatment of FRSS issues. No specific discussions of seismically induced failure of fire suppression systems were provided in the submittal.

b. Seismic-Fire Walkdown

The submittal does not indicate that a seismic-fire walkdown evaluation was conducted.

c. Seismically Induced Flood Evaluation

No documentation pertaining to evaluation of seismically induced floods was submitted.

d. Review Findings

The Turkey Point seismic adequacy evaluation has not fully addressed seismic-fire interactions or seismically induced floods.

2.1.13 Treatment of USI A-45

A reduced-scope seismic assessment should consider the seismic capability of components necessary for successful decay heat removal, in response to USI A-45 (Decay Heat Removal Requirements).

FPL's seismic IPEEE submittal and seismic adequacy evaluation study for Turkey Point did not directly document findings for any generic issues (GIs)/USIs other than USI A-46. Indirectly, USI A-45 was partially addressed owing to the fact that the success path needed to accomplish one method of decay heat removal (i.e., via the AFW system). However, the AFW pumps were eliminated from the seismic evaluation (because they had been previously examined for seismic adequacy elsewhere), and only the condensate storage tanks (CSTs) were identified as necessary components in the SSEL.

The licensee's seismic IPEEE submittal does not address a meaningful scope of components related to decay heat removal functions. This weakness stems from the fact (noted in Section 2.1.2 of this TER) that the SSEL is only partially complete.

2.1.14 Treatment of GI-131

GI-131, "Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants," is applicable to Turkey Point Units 3 and 4. FPL has stated that lateral restraint was added to the movable support assembly of the flux mapping system in 1989. This additional lateral restraint was analyzed by Westinghouse, Bechtel, and FPL, and determined to be adequate to prevent any loss of reactor coolant boundary due to a design-basis earthquake at Turkey Point. The licensee states that this issue had been discussed and closed by the NRC in Turkey Point Inspection Report IR-92-16.

GI-131 appears to have been satisfactorily resolved based on earlier NRC acceptance. The SRT did not perform an evaluation of the in-core flux mapping system as part of the USI A-46/IPEEE seismic adequacy study of Turkey Point Nuclear Plant.

2.1.15 Peer Review Process

An independent external peer review was conducted by Dr. Paul Smith for the seismic adequacy evaluation study of Turkey Point Units 3 and 4. No additional seismic concerns were identified from the review. FPL engineers also reviewed the seismic studies.

A meaningful peer review appears to have been conducted for the limited-scope seismic evaluation study of Turkey Point Nuclear Plant.

2.1.16 Summary Evaluation of Key Insights

Only a subset of components needed to assure successful shutdown are considered in FPL's equipment list, and hence, the seismic IPEEE process has only a limited potential to reveal vulnerabilities or outliers. However, for those components that have been included in the scope of FPL's seismic adequacy evaluation study, the process implemented for screening outliers, and for addressing their resolution, is considered to be appropriate and adequate.

FPL's seismic adequacy evaluation study has identified a significant number (35) of outliers (primarily relating to weak anchorage), and has proposed relevant modifications to enhance safety. The NRC has already reviewed these outliers and modifications for Turkey Point, Units 3 and 4, as part of USI A-46 resolution. Additionally, the NRC has conducted a site investigation to identify any vulnerabilities that may require further analysis/treatment. As a result, FPL is performing follow-up actions to resolve concerns related to: an anchorage problem, a maintenance issue related to corrosion, a potential seismic interaction, and the need for a strict housekeeping program.

Even after proposed plant safety enhancements are implemented, the upgraded HCLPF capacity of the condensate storage tanks and refueling water storage tanks, as reported in the seismic IPEEE submittal, is still only 0.11g, which is less than the design-basis acceleration of 0.15g. However, in its USI A-46 review, the NRC determined the HCLPF capacity of the tanks to be in excess of 0.12g PGA, and sufficiently close to the design level [7].

No other outliers reported by the licensee appear to require further analysis for seismic IPEEE purposes. However, additional outliers may have well been found if the licensee had expanded the scope of its seismic adequacy evaluation to address IPEEE-only components and issues. Furthermore, the licensee elected not to conduct a containment performance analysis at Turkey Point Nuclear Plant. Thus, no vulnerabilities affecting containment performance, related to seismic behavior of containment systems (e.g., containment cooling, containment isolation, etc.), nor pertaining to direct seismic failure of the containment structures themselves, were identified.

The Turkey Point Nuclear Plant seismic adequacy evaluation study is capable of finding only a limited set of seismic-related, severe accident vulnerabilities. Low HCLPF capacities have been noted for some essential storage tanks, but were accepted by the NRC as part of USI A-46 resolution.

2.2 Fire

A summary of the licensee's fire IPEEE process has been described in Section 1.2. Here, the licensee's fire evaluation is described in detail, and discussion is provided regarding significant observations encountered in the present review.

2.2.1 Overview and Relevance of the Fire IPEEE Process

a. Method Selected for Fire IPEEE

The fire analysis was conducted using the 1990 version of the fire-induced vulnerability evaluation (FIVE) methodology [19], including all three phases of the methodology. The first phase is a screening step, and is based primarily on the contents of a fire area. In the second phase, the frequency of core damage due

to fires in a specific fire zone is estimated using the formulations and data provided as part of the FIVE methodology. A fire area, per the licensee's definition, may contain several fire zones. The third phase of FIVE application involves plant walkdowns and verification of results of the first two phases; it also involves additional modeling of the chain of fire events, i.e., of fire phenomenology. Six fire zones were addressed in detail. For two of these zones (zones 119 and 120), results of fire modeling have been discussed. For the other four fire zones (the control room, cable spreading room and zones 63 and 61), an analysis of the chain of events (i.e., internal events equipment failures), and of operator actions, has been considered.

b. Key Assumptions Used in Performing Fire IPEEE

The IPEEE submittal does not provide a separate list of analysis assumptions. However, the present review has identified the following assumptions which could have a significant influence on the final results:

- Reactor subcriticality was assumed to be successful in all cases.
- The possibility of using feed and bleed was not considered for any of the fire scenarios.
- Containment fires were not analyzed explicitly. The apparent basis for this approach is that earlier PRAs have generally found the contribution of containment fires, to overall fire risk, to be negligible.
- Reported fire frequencies for control rooms and cable spreading rooms are very conservative.
- Fire barriers/boundaries were taken to be as good as rated. Per Reference [13], because of the established procedures for inspection and maintenance, the failure possibility of active fire barriers (i.e., open doors, ducts, dampers, etc.) was assumed to be negligible. This assumption has resulted in the conclusion that cross-zone fires are negligible contributors to risk.

c. Status of Appendix R Modifications

The licensee has not indicated the status of Appendix R modifications. Information and procedures generated as a result of the Appendix R effort were used as the basis for the fire analysis.

d. New or Existing PRA

The fire IPEEE was derived from an existing study. It used the results of an already completed IPE and PRA for Turkey Point Nuclear Plant, Units 3 and 4 [11].

2.2.2 Review of Plant Information and Walkdown

a. Walkdown Team Composition

Two PRA engineers and a member of plant fire protection personnel have conducted the walkdown. No details are provided as to how the walkdown was conducted and its duration. All areas of the plant outside the containment have been visited. The main focus of the walkdown is claimed to be on the verification

of the location of safe shutdown equipment. It must be noted that the main concern in fire risk analyses is the location of cables of redundant trains. Identification of the position of a specific cable within a compartment is practically impossible unless that cable is traced with special tracing mechanisms. This, of course, could not have been done at this walkdown.

b. Significant Walkdown Findings

References [1], [11] and [13] do not indicate that the walkdown team discovered any new fire vulnerabilities from the plant fire walkdown.

c. Significant Plant Features

The following is a list of plant features that are deemed to be important:

1. The emergency steam-driven main feedwater pump can be used for secondary heat removal.

2. There are several safe shutdown related equipment items and cabinets that are not located within structures, but are openly exposed to environmental elements. This configuration is mentioned as an important feature of Turkey Point, Units 3 and 4, that eliminates the concern for compartment fires and formation of hot gas layers.

3. Several safety systems are shared between the two units.

4. There is one control room and one cable spreading room for both units.

5. Combined loss of the CCW system and of charging pump seal injection will lead to RCP seal failure.

2.2.3 Fire-Induced Initiating Events

a. Were Initiating Events Other than Reactor Trip Considered?

There is no explicit and focused discussion in the IPE [11] on the possibility of a fire leading to the initiating events considered in the IPE. However, it is emphasized in Reference [13], that the licensee has considered the possibilities of occurrence of loss of offsite power, reactor coolant pump seal failure, inadvertent opening of a power-operated relief valve (PORV), and failure of the ICW and CCW systems, from a fire-induced event.

b. Were the Initiating Events Analyzed Properly?

A functional fault tree was developed to account for possible fire-induced initiating events and the possibility of reactor trip, LOCA, loss of offsite power, etc. Based on the discussions provided in References [11] and [13], it can be concluded that the issue of fire-induced initiating events has been properly considered.

2.2.4 Screening of Fire Zones

a. Was a Proper Screening Methodology Employed?

Screening was accomplished following the FIVE methodology. A list of all fire areas and zones is provided in Reference [11], along with the equipment that could potentially be affected by a fire in each area, as well as the specific screening criteria and assumptions associated with the fire areas and zones.

In Section 3.7.1 of the Turkey Point PRA, it is stated that a fire area was screened out if it does not lead to a reactor trip. This approach, in general, is not conservative. However, since an area would not be screened out if it contains at least one item of safe shutdown equipment (the reviewers assume that equipment also includes cables), the submittal's treatment related to reactor trip has a minimal impact on the fire risk evaluation.

b. Have the Cable Spreading Room and Control Room Been Screened Out?

The cable spreading room and control room have been included in a detailed analysis, and were not screened out in Phases I and II. However, the discussion related to the detailed analyses for these rooms is cursory. It is claimed that fire initiation frequencies used in the industry are conservative. No discussion is provided as to whether there are any formal methods for switching the controls to the alternate shutdown panel, as to what functions can be controlled and monitored from the panel, and as to how many operators are required, etc. Furthermore, the licensee does not indicate whether there are any cables that are grouped together in the cable spreading room, which can cause damage to a critical set of equipment. If such a grouping exists, the argument of low frequency for cable damage is weakened. Also, there are no indications in the submittal regarding whether there are any power cables in the cable spreading room.

c. Were There Any Fire Zones/Areas that Have Been Improperly Screened Out?

The results of the screening activities are presented in a tabular form. Based on information provided in References [11] and [13], for zones 32, 33, 47, 51, 121, 122, 124, 133, 135, and 139, the critical cables are either embedded in concrete or are located underground below the fire zone. For fire zones 12, 41, 54, 62, 76, 78, and 79, a sub-set of the critical cables are embedded or are located underground. However, the rest of the cables are exposed (unprotected) and, in some cases, manual actions have to be undertaken to properly align plant systems for core cooling function.

The licensee provides a list of fire zones that have been screened out based on the frequency of core damage (less than 10^{-6} per reactor-year). The fire zones that have been selected for detailed analysis (the fire zones that could not be screened out), with the exception of the control room and the cable spreading room, do not typically appear within the list of significant fire risk areas for other PWRs.

Manual actions may need to be undertaken to assure availability of a redundant train. There are no indications as to whether the effects of a specific fire on such actions has been considered in the analysis. Fire zone 79A is a cable riser having a large array of cables in it. The licensee does not indicate whether there are any power cables in this riser. The area was screened using Phase II analysis. That is, the frequency of fire damage was established for the area, and was found to be less than 10^{-6}/yr. Given that

a fire in such a compartment can cause severe damage to disabled equipment, it is important for the licensee to show the basis for reaching the screening conclusions.

2.2.5 Fire Hazard Analysis

FIVE methodology and data bases were used. The overall results and frequency values are within the range of frequencies considered for other PWRs. Except for the following four areas, these frequencies are conservative because the equipment within the screened out areas are not included in the data base for partitioning the overall fire frequency of the plant. The fire frequencies for fire zones 6 and 7 (gas compressor rooms), 128 (the switchyard) and 132 (control room electrical chase) could be low. Fire zone 132, the control room electrical chase, is the only zone for which the choice of low frequency may have led the licensee to overlook a potential vulnerability.

A plant-specific data base has not been used. The use of generic data base could be viewed as optimistic, if Turkey Point Units 3 and 4 have experienced fire events in safety-related areas.

2.2.6 Fire Growth and Propagation

Fire growth and propagation have been modeled in a limited number of fire zones. The submittal presents only one case (the ICW pump area) of fire propagation analysis where the possibility of oil spill and damage to adjacent pumps have been modeled. The formulations provided in FIVE have been used for this purpose.

a. Treatment of Cross-Zone Fire Spread and Associated Major Assumptions

It is claimed that the fire compartment interaction analysis (FCIA) methodology described in FIVE [19] has been used. The possibility of active fire barrier failure has not been analyzed assuming that the current procedures for maintaining these barriers ensures a high component reliability. Reliance solely on fire barrier effectiveness where there are active barriers could be optimistic.

b. Assumptions Associated with Detection and Suppression

The specific fire detection and suppression characteristics of various fire zones have not been addressed and analyzed. The IPEEE submittal claims, that except for the control room, no credit was taken for manual fire suppression. Suppression system failure probability is mentioned in Reference [11] for the cable spreading room, only. A simple model was used for this purpose. There are no discussions regarding the competing phenomena of fire damage, and fire detection and suppression.

c. Treatment of Suppression-Induced Damage to Equipment, if Applicable

It is claimed that the fire analysis was performed in accordance with the guidance of the FIVE methodology, which does not address fire suppression system induced damage. This omission can be optimistic if it is totally ignored from the analysis.

d. Computer Codes Used, If Applicable

There is no mention of any specific computer program used in the fire analysis.

2.2.7 Evaluation of Component Fragilities and Failure Modes

a. Definition of Fire-Induced Failures

It is inferred that fire-induced failures have been considered properly. In Reference [11], it is discussed that spurious actuation of valves was considered in the analysis. This failure mode was included even in those scenarios where the power source to the valve may be lost as a result of the fire event. However, no concise and specific discussion has been provided for component fragilities and failure modes.

b. Method Used to Determine Component Capacities

No criteria is mentioned regarding survival capacities of cables and electrical equipment. Given the vintage of the plant, the fact that FIVE methodology was used, and that Appendix R requirements have been met, the licensee is expected to have used the proper failure criteria.

c. Generic Fragilities

No specific discussion is provided regarding equipment and cable fragilities. However, from the discussions provided in Reference [11], it can be inferred that hot shorts and other failure mechanisms have been considered.

d. Plant-Specific Fragilities

Plant-specific fragilities have not been used.

e. Technique Used to Treat Operator Recovery Actions

Operator recovery actions were addressed in several cases. For recovery from control room or cable spreading room fires, however, the licensee provides a discussion that fire and smoke will not interfere with the operator actions to reach and manipulate the alternate shutdown panel. The following is a list of operator actions mentioned in Reference [11]:

- Operator actions are required to operate valve hand-wheels in case of a fire in the reactor control rod equipment room.
- Operator actions are required to cross-tie the CCW system, or to connect a hose to the service water system for the charging pump oil cooler, in order to establish RCP seal integrity in case of loss of ICW.

A simple model was employed for human recovery actions in the case of a control room or a cable spreading room fire.

2.2.8 Fire Detection and Suppression

The licensee claims that the methodology and data provided in FIVE have been used to model the effects of fire detection and suppression systems. However, fire detection and suppression, as discussed above, are not mentioned for almost all of the areas and zones, except for the cable spreading room. A system

unavailability of 0.05 was used for the Halon system. This treatment could be optimistic if a critical set of cables and equipment are within a small part of the room. In other words, regardless of failure or success of fire detection and suppression, a critical set of cables can be so close together that, in case of a fire occurring within that specific area, the equipment and cables would be rendered failed by the fire before the suppression system has an opportunity to stop the damage.

2.2.9 Analysis of Plant Systems and Sequences

a. Key Assumptions Including Success Criteria and Associated Bases

The success criteria were directly taken from Reference [11], the IPE.

b. Event Trees (Functional or Systemic)

The sections reviewed for fire analysis do not distinguish between functional and systemic event trees.

c. Dependency Matrix, if it is Different from that for Seismic Events

A dependency matrix was not provided in the sections reviewed for fire analysis.

d. Plant-Unique System Dependencies

The submittal does not identify any plant-unique system dependencies, relevant to fire risk, except for indicating that Units 3 and 4 share several components that can be used to provide cooling or power from one unit to the other.

e. Shared Systems for Multi-Unit Plant

There are several systems and areas (fire zones) shared between the two units: auxiliary feedwater system; high head safety injection; portions of the electric power system; the chemical and volume control system; structures (including the cable spreading room and the control room); and portions of the HVAC system. Several systems can also be cross-tied between the units, including component cooling water, main feedwater, instrument air, and AC power.

Recovery actions using a cross-tie were considered for the loss of ICW analysis.

f. Most Significant Human Actions

The IPEEE has not addressed human actions as a separate item. For all six unscreened areas, human actions are critical to plant safety. In the cases of the cable spreading room and control room, a very simple model (probability of 0.1) was used to account for the possibility of operator failure in controlling the plant from outside the control room using the alternate shutdown panel. It is stated by the licensee that none of the fires that may require the activation of the alternate shutdown panel would lead to a condition that could potentially hinder the alternate shutdown panel activation.

For the four risk-significant areas, other than the control room and cable spreading room, the actions are specified, but no probabilistic analysis is provided. The following is a list of fire zones and associated human actions:

- Cable Spreading Room: Operators switch the controls of one train to the alternate shutdown panel, and take control of the plant using the alternate shutdown panel.

- Control Room: The operator actions are the same as those for the cable spreading room.

- Zone 63, Reactor Control Rod Equipment Room, Unit 3: Operators should re-establish RCP seal cooling or injection by locally opening specific valves manually, using their respective hand-wheels. It is stated that these actions can be taken from an area that would not be affected by the fire.

- Zone 61, Reactor Control Rod Equipment Room, Unit 4: Operator actions are the same as those for Zone 63; and similar to the action for zone 63, it is stated that the operator actions can be taken from an area that would not be affected by the fire.

- Zone 119, Unit 3 Intake Structure: Operators should either cross-tie CCW to the other unit, or connect service water to the "B" charging pump oil cooler.

- Zone 120, Unit 4 Intake Structure: Operator actions are the same as those for Zone 119.

The licensee does not provide any additional information as to how failure of the operators to properly carry out the required actions has been modeled. It is stated that since these issues have not been addressed in FIVE [19], the licensee has chosen not to address them in the submittal.

2.2.10 Core Damage Frequency Evaluation

The licensee has provided some information regarding lost functions and the fire-induced initiating events. The licensee claims that the frequencies used in the screening phases cannot be considered as representing core damage, since only two functions - secondary heat removal and reactor coolant system integrity - have been considered in the analysis.

2.2.11 Analysis of Containment Performance

a. Significant Containment Performance Insights

Containment fires were not included, and containment isolation failure was not explicitly addressed in the IPEEE.

b. Plant-Unique Phenomenology Considered

No containment related event trees have been used in any of the screening phases, nor in evaluating the unscreened fire zones.

2.2.12 Treatment of Fire Risk Scoping Study Issues

a. Assumptions Used to Address Fire Risk Scoping Study Issues

1. Seismic-fire interaction was addressed through the failure of fire suppression systems and their effect on safety equipment. It is stated that fire suppression design, for areas where it can affect safe shutdown equipment, includes provisions to minimize inadvertent actuation from a seismic event.

   The possibility of fire occurrence from seismic activity is addressed through the control of the installation of "temporary" components. For example, proper chaining of gas cylinders and temporary equipment are mentioned as counter-measures for minimizing the possibility of seismically induced fire.

   No further information is provided regarding seismic-fire interaction. As it is discussed in Section 2.1.12 of this report, several important issues have not been addressed by the licensee. It is concluded in Section 2.1.12 that seismic-fire interactions were not addressed adequately.

2. Specific procedures were cited for the design, inspection and maintenance of the fire doors, fire dampers, fire barriers and penetration seal assemblies.

3. Several procedures are implemented for fire detection, fire fighting, general personnel training, and fire brigade training and drills. The IPEEE states that all plant personnel, who have unescorted access, must undergo fire watch training. In addition, strict training and drills are required for the fire brigade. Fire extinguishers are provided throughout the plant.

4. No discussion is provided on the potential for adverse effects of the actuation of a fire suppression system on safety systems, nor concerning the potential impact of combustion products on equipment. Operator performance issues have been addressed. Specific procedures have been cited for safe shutdown in case of a fire, for operator training, and for fire training.

5. Control system interaction is addressed via the use of an alternate shutdown panel and isolation switches. A simple model was used for the operators failing to control the plant from this panel in case of a control room or cable spreading room fire. References [1] and [11] do not provide a list of equipment and instrumentation that can be controlled and monitored from the alternate shutdown panel. Reference [13] indicates that the analysts have considered the possibility of a fire hindering access to the isolation switches or alternate shutdown panels.

b. Significant Findings

1. The fire brigade undergoes sufficient training, and all personnel who have unescorted access act as fire-watch.

2. The suppression systems, in safety-related areas, can withstand seismic events; therefore, seismically induced failure of fire equipment is not a concern.

3. Procedures are available that address fire-related issues.

4. Potential adverse effects on plant equipment, by spurious suppression system actuation, and by combustion products, were not addressed.

5. No consideration was given to mechanical failure of active barriers.

2.2.13 USI A-45 Issue

Turkey Point Nuclear Plant has been used as a case study plant by Sandia National Laboratories for probabilistic evaluation of decay heat removal adequacy, in the context of USI A-45. This issue was addressed as part of the IPE in general (which includes the fire analysis part of the IPEEE).

a. Methods of Removing Decay Heat

For fire analysis, only secondary side heat removal via the auxiliary feedwater system has been included in the models.

b. Ability of the Plant to Feed and Bleed

Turkey Point, Units 3 and 4, have this capability; however, this has not been mentioned in the fire analysis.

c. Credit Taken for Feed and Bleed

No credit has been taken for feed and bleed capability.

d. Presence of Thermo-Lag

Although references [1] and [11] do not mention the presence of Thermo-Lag at Turkey Point Nuclear Plant, Units 3 and 4, Thermo-Lag is present at this plant. The resolution of the Thermo-Lag issue may have a profound impact on the results of the IPEEE fire analysis.

2.3 <u>HFO Events</u>

The IPEEE submittal reports that there are no vulnerabilities to severe accident risk from external events (Section 1.4 of Reference [1]). The HFO events which are explicitly examined include hurricanes, tornadoes, external flooding, transportation and nearby facility accidents, and lightning. The dominant risk, however, is shown to be that associated with storm surge flooding that may be accompanied by a beyond design basis hurricane (Page 8 of Reference [1]).

Because Turkey Point, Units 3 and 4, were constructed prior to the issuance of the 1975 SRP, the methods in NUREG-1407 [3] for non-SRP plants were used. The general methodology used in the study follows that presented in NUREG-1407 for the analysis of other external events. The following are the major steps in the approach:

1. Establish a list of plant-specific other external events,

2. Perform progressive screening, and

3. Document approach and findings.

The progressive screening step includes the following items:

- Review of plant-specific hazard data and licensing bases
- Identification of significant changes since the plant operating license (OL) was issued
- Establishing whether the plant and the facilities design comply with the 1975 SRP criteria
- Determining whether the hazard frequency is acceptably low
- Performing a bounding analysis, if necessary
- Performing a Probabilistic Risk Assessment (PRA), if necessary

In addition to the hazards already mentioned, the IPEEE submittal considered a number of other hazards that could apply to the Turkey Point site. Table 5-19 of the submittal lists the other hazards considered and gives a brief evaluation of each hazard. In addition, NUREG/CR-4762 [20] was referenced as an information source that was used to gain insights on potential hazards to the plant.

The following subsections provide a summary of the analysis performed for each major hazard.

2.3.1 High Winds and Tornadoes

2.3.1.1 General Methodology

Turkey Point, Units 3 and 4, were designed and built prior to the NRC's current licensing criteria. Thus, the NUREG-1407 approach for non-SRP plants has been used for a systematic examination approach for the IPEEE. The steps that were utilized included: a bounding analysis, a determination that the hazard frequency was acceptably low, or a PRA assessment, as allowed by NUREG-1407.

2.3.1.2 Plant-Specific Hazard Data and Licensing Basis

The IPEEE submittal for Turkey Point, Units 3 and 4, analyzed high winds from two sources: hurricanes and tornadoes. The former analysis was presented in the Turkey Point, Units 3 and 4, IPE submittal (Section 3.8 of Reference [11]). The latter is contained in the IPEEE submittal itself.

The hurricane analysis divided the hazard into four main components requiring evaluation: wind, tidal surge, wind-generated missiles, and precipitation.

To estimate the wind speed hazard, several studies were reviewed for applicability to Turkey Point. NUREG/CR-4762 (Reference [20]) determined the equipment most susceptible to high wind. The Units 1 and 2 (fossil unit) stacks were determined to be susceptible to failure as a result of high winds. All other Class 1 structures have been designed for wind speeds greater than those that would be expected from a hurricane. The stack failure frequency was re-calculated, and the IPE PRA model was then used to

generate accident sequences based upon equipment that would be failed due to a collapsed stack. In all cases, the sequence frequencies were found to be less than 10^{-7}/ry.

To estimate the maximum storm surge, a numerical model (SLOSH) developed by the National Hurricane Center was used to provide a relationship between surge height and storm strength. The analysis of wind velocity and recurrence performed by the University of Florida and the National Hurricane Center provided two boundaries on wind speed. Together, these sets of data showed that the maximum expected storm surge would be approximately 18 feet. This storm surge would overtop the plant flood-wall and thus inundate critical safety-related equipment (emergency switchgear). The frequency for such a storm surge was estimated to have an upper bound value of 10^{-4}/ry and a lower bound value of 10^{-6}/ry. This result is one of the important findings of the submittal.

Hurricane-generated missiles were evaluated, assuming a maximum wind speed of 200 mph, and using a tornado missile simulation code developed by EPRI, called TORMIS. The SRP missile spectrum was used, and equipment susceptibility values were taken from the Oconee PRA (Reference [21]). The internal events plant PRA model was then used to evaluate a resulting core damage frequency. All resulting sequence frequencies were found to be below 10^{-7}/ry.

The impact of hurricane-generated precipitation was also evaluated in the IPEEE/IPE. It was stated that the control building was evaluated as being able to withstand an accumulation of water of up to 6 inches. Other buildings do not have sills, so the accumulation of water was stated to be insignificant. Condensate pit flooding was also considered, but was shown to have a frequency of less than 10^{-6}/ry. Lastly, the plant PRA model was evaluated using high conditional failure probabilities for outdoor equipment. The CDF results were found to be bounded by the storm surge results.

The impact of tornado hazard was evaluated considering contributions from: (1) a high wind component, and (2) a missile component. A family of mean hazard curves for wind speed had been developed specifically for Turkey Point in NUREG-4762 [20]. The assessment of the missile component of tornado hazard relied upon the Turkey Point PRA hurricane missile analysis. It is unclear, however, whether or not the TORMIS missile simulation code used in the PRA for the hurricane analysis was re-executed for wind speeds greater than 200 mph (Page 3.0-283 Reference [11]). The SRP missile spectrum was used, and the fraction of failures for missile impact, conditional upon the occurrence of a tornado, was developed for several major components, including the Units 1 and 2 fossil stack (Table 5-9 Reference [1]). These components were then added to the internal events plant PRA model as new basic events, and the model was re-analyzed. The resulting CDF due to combined tornado wind and missile was found to be 4.81×10^{-7}/ry for Unit 3, and 6.85×10^{-7}/ry for Unit 4.

2.3.1.3 Significant Changes Since Issuance of the Plant Operating License

The submittal does not identify any significant changes that have occurred since the time the OL was issued.

2.3.1.4 Significant Findings and Plant-Unique Features

Per Reference [13], the major walkdown findings related to high winds and tornadoes are as follows:

1. Determined that the flood wall represented the minimum vulnerability height for the power block;

2. Verified that the plant layout at the time of the walkdown agreed with drawings used in the analysis and confirmed the relationship of the equipment in the plant to flood elevation as well as potential exposure to wind, missile, and precipitation;

3. Identified the surge hazard associated with storm to be the most important component of the hurricane hazard; and

4. Determined that the Unit 2 stack was vulnerable to high winds. (The Unit 2 stack was damaged during Hurricane Andrew and was replaced with a stronger version in 1993.)

A summary of the walkdown procedures used by the licensee, and a description of the qualification of the team members performing the walkdown, are not provided in the submittal.

2.3.1.5 Hazard Frequency

The high-wind hazard frequency was taken from the National Hurricane Center HURISK model, which is presented in the Turkey Point PRA. The tornado hazard curves were developed specifically for Turkey Point in NUREG-4762 [20] by Sandia National Laboratories. These data were then applied to the internal events plant PRA model for further evaluation.

2.3.1.6 Bounding Analysis

The results of the bounding analysis indicate that high winds could create a storm surge which could overtop the plant flood walls, and thus, inundate critical safety equipment. An upper-bound recurrence frequency of 10^{-4}/calendar-year and a lower-bound recurrence frequency of 10^{-6}/calendar-year were estimated. This estimated bound on recurrence frequency was performed using the National Hurricane Center's SLOSH model.

2.3.1.7 PRA Analysis

In the tornado analysis, the hazard curves were developed specifically for Turkey Point in NUREG-4762 [20]. The assessment of the missile hazard and fragility relied upon the Turkey Point PRA hurricane missile analysis. New basic events were added to the plant PRA model to represent components susceptible to high winds (e.g., refueling water storage tank (RWST), fossil stack, and condensate storage tank). The resulting CDF due to tornado wind and missile was found to be 4.81×10^{-7}/ry for Unit 3, and 6.85×10^{-7}/ry for Unit 4.

2.3.2 External Flooding

2.3.2.1 General Methodology

The methodology for external flood analysis consists of first determining the credible flooding sources. For those sources found credible, the plant's minimum levels for flood propagation pathways, and the maximum possible external flooding levels, are determined. If the plant elevation precludes flooding from these maximum flooding levels, the analysis is complete. If the plant elevation does not preclude flooding, further analysis is required.

2.3.2.2 Plant-Specific Hazard Data and Licensing Basis

The Turkey Point Nuclear Plant flood wall requires that a flood exceed 18 ft above mean low water (MLW) before plant safety-related equipment is damaged, due to the existence of the flood wall, and since the elevation of most of the surrounding area is less than 18 ft (Page 22 Reference [1]). Consequently, the most credible source of external flooding cited is due to storm surge.

The Turkey Point, Units 3 and 4, PRA (Reference [11]) was referenced as the source of the flooding analysis. As a component of the hurricane analysis found therein, storm surge as a source of flooding was studied. Other sources of flooding were screened out as shown in Table 5-19 of the IPEEE submittal.

To estimate the maximum storm surge, a numerical model (SLOSH) developed by the National Hurricane Center provided a relationship of surge height and storm wind velocity (Figure 3.8-11 of Reference [1]). The analysis of wind velocity and recurrence performed by the University of Florida and the National Hurricane Center provided two boundaries for wind speed. Together, these sets of data showed that the maximum expected storm surge would be approximately 18 feet. This storm surge would overtop the plant flood wall and inundate critical safety-related equipment (emergency switchgear). The frequency was estimated to have an upper bound of 10^{-4}/ry and a lower bound of 10^{-6}/ry. This is one of the important findings of the submittal.

The submittal uses a Danish Hydrologic Institute study in assessing the impact of tide and waves on storm surges. Table 3.8-4 in the IPE submittal (Reference [11]) estimates the wave size to range from 3.9 to 9.6 feet.

With respect to the effects of intense rainfall on roof loads, the submittal notes that the control building can withstand water accumulation up to 6 inches; above the level of 6 inches water spills over the sides of the roof. Increases in rainfall intensity would only result in a more rapid accumulation of water on the roof. The submittal also mentions that the roofs of other buildings at the plant would not accumulate water.

2.3.2.3 Significant Changes Since Issuance of the Plant Operating License

The submittal does not identify any significant changes that have occurred since the time the OL was issued.

2.3.2.4 Significant Findings and Plant-Unique Features

Storm surge was found to dominate all other hurricane, tornado, and external flooding hazards. This conclusion was drawn in the IPE submittal (Page 3.0-287 Reference [11]).. As a consequence, FPL chose to take actions to minimize the risk posed by beyond design basis hurricanes. Section 6 of the IPE describes the actions to be taken.

A summary of the walkdown procedures employed by the licensee, and the qualification of the team members performing the walkdown, are not provided in the submittal.

2.3.2.5 Hazard Frequency

The external flooding hazard was developed in the IPE submittal during the analysis of hurricane events. The frequency of flooding was then related to the frequency of occurrence of a beyond design basis hurricane (see Section 2.3.1.5).

2.3.2.6 Bounding Analysis

The results of the bounding analysis indicate that external flooding, caused by a storm surge associated with a beyond design basis hurricane, could overtop the plants flood walls, and thus, inundate critical safety-related equipment. Such a level of storm surge was estimated to have an upper-bound recurrence frequency of about 10^{-4}/calendar-year and a lower-bound recurrence frequency of about 10^{-6}/calendar-year. Section 6 of the IPE addresses the actions to be taken to reduce the risk of this hazard.

2.3.3 Transportation and Nearby Facility Accidents

2.3.3.1 General Methodology

On page 26 of the submittal, it is stated that the methodology uses a progressive screening approach that is described in NUREG-1407. The submittal assessment determines whether the plant meets the SRP with respect to each identified hazard. For those hazards found to have met the criteria, no further analysis was performed. FPL has determined that aircraft crashes, toxic chemical releases and explosions do not meet the SRP criteria, and thus, are analyzed in more detail. Table 5-15 of the IPEEE submittal summarizes the criteria used for each hazard. Most hazards have been eliminated due to the distance of the plant site from the hazard source.

2.3.3.2 Plant-Specific Hazard Data and Licensing Basis

The screening criteria, as well as the methodology, in the SRP is cited as the source for the analysis of aircraft crashes from all credible causes. In addition, FPL used an aircraft crash methodology developed by Sandia National Laboratories to compare with the SRP methodology. The result of both methods showed the risk from this hazard as insignificant (8.5×10^{-7}/year is the largest value for crash frequency that was reported for any airport; see Table 5-17 of Reference [1]).

Four sources of explosion hazard were analyzed: an onsite liquid propane gas (LPG) tank, onsite natural gas storage, an onsite hydrogen trailer, and jet fuel storage at nearby Homestead Air Force base. These sources were derived from a plant walkdown and discussion with plant personnel. However, no procedures for the walkdowns or discussions are provided. The accuracy of the methodology and assessment cannot be ascertained. Moreover, documentation is lacking as to whether or not the Air Force base was contacted to determine if there are additional explosive hazards present, other than just jet fuel.

The LPG tank explosive hazard was screened out based upon SRP criteria.. The analysis of the natural gas explosion hazard determined that the intake structure is the plant structure that would be most susceptible to such an explosion. The estimate of core damage frequency from the failure of the intake structure due to natural gas explosion was placed at 6.0×10^{-8}/ry, and could, therefore, be screened out.

The submittal indicates that the Florida Gas Transmission Company expects to isolate a leak within 15 to 20 minutes following a rupture. It is not clear whether this figure includes the amount of time that elapses before the leak is discovered. This will affect the assumed 0.05 probability of isolating the rupture within 30 minutes.

The hydrogen trailer, as an explosion source, marginally exceeded the SRP criteria. Further scoping analysis indicated that the most susceptible components are the balance of plant (BOP) equipment outside the turbine structure, and one train of 4kV switchgear. The core damage frequency was estimated to be less than 10^{-7}/ry. This result was based, however, on an estimated initiating event frequency of 10^{-3}/year, for which there was no basis cited.

Homestead Air Force base, as a source of jet fuel explosions, was screened out because it is located more than 5 miles from the plant.

Toxic chemical releases were also analyzed, and only four chemicals were found in sufficient quantities (greater than 100-pound containers) to be considered further: chlorine, sulfuric acid, sodium hydroxide, and hydrazine (Table 5-18 of Reference [1]). Chlorine is stored in such small quantities that it meets Regulatory Guide 1.95 (Reference [22]) and is, therefore, eliminated. The submittal states that the sulfuric acid and sodium hydroxide tanks are empty and no longer used; however, a discussion of the justification for eliminating sulfuric acid is still presented. The analysis of a hydrazine spill indicates that the vaporization rate fails to develop a concentration in the control room ventilation at a level above the level of concern (LOC). No further analysis was determined to be required.

2.3.3.3 Significant Changes Since Issuance of the Plant Operating License

The submittal does not identify any significant changes since the time the OL was issued.

2.3.3.4 Significant Findings and Plant-Unique Features

Per Reference [13], the walkdown did not identify any potential vulnerabilities associated with the storage of hazardous materials, or transportation or nearby facility accidents. A summary of the walkdown procedures used by the licensee, and the qualification of the team members performing the walkdown, are not provided in the submittal.

2.3.3.5 Hazard Frequency

Air transportation accidents are screened out by hazard frequency. The most conservative value presented (for a given airport) is 8.5×10^{-7}/year. The estimate of the hazard frequency due to natural gas explosion was placed at 10^{-4}/year, and that due to hydrogen explosion at 10^{-3}/year. All other transportation accidents or nearby facility accidents were deterministically screened out; therefore, no hazard frequencies were reported.

2.3.3.6 Bounding Analysis

The estimated core damage frequency, as a result of a natural gas explosion in the intake structure, is placed at 6.0×10^{-8}/ry and could, therefore, be screened out. Core damage as a result of a hydrogen explosion was estimated to occur with a frequency less than 10^{-7}/ry.

2.3.4 Lightning and Others

A discussion of lightning is presented in Section 5.4.1 of the IPEEE submittal. The Turkey Point Nuclear Plant site has experienced two lightning events in its operating history, both of which caused Unit 3 to trip. No visual damage occurred, and it was determined that the components which initiated the spurious trip were overly sensitive to lightning, and consequently, they were recalibrated. Also presented in the IPEEE is Sandia National Laboratory's (SNL) calculated core damage frequency contribution from lightning. The result was determined as 2×10^{-6}/ry per unit, but was calculated prior to the electrical system upgrade and the addition of two emergency diesel generators. FPL concluded that there is no unique vulnerability to lightning at Turkey Point.

Table 5-19 of the submittal lists other events that were considered in the review, together with a statement of their applicability to Turkey Point.

2.4 Generic Safety Issues (GSI-147, GSI-148, GSI-156 and GSI-172)

2.4.1 GSI-147, "Fire-Induced Alternate Shutdown/Control Panel Interaction"

GSI-147 addresses the scenario of fire occurring in a plant (e.g., in the control room), and conditions which could develop that may create a number of potential control system vulnerabilities. Control system interactions can impact plant risk in the following ways:

- Electrical independence of remote shutdown control systems
- Loss of control power before transfer
- Total loss of system function
- Spurious actuation of components

The licensee, as indicated on page 4 of Reference [13], has considered the possibilities of occurrence of LOCAs, loss of offsite power, reactor coolant pump seal failure, inadvertent opening of a PORV, and failure of the ICW and CCW systems, from a fire-induced event. Since the submittal has followed the guidance provided in FIVE concerning control system interactions, all circuitry associated with remote shutdown is assumed to have been found to be electrically independent of the control room.

2.4.2 GSI-148, "Smoke Control and Manual Fire Fighting Effectiveness"

GSI-148 addresses the effectiveness of manual fire-fighting in the presence of smoke. Smoke can impact plant risk in the following ways:

- By reducing manual fire-fighting effectiveness and causing misdirected suppression efforts
- By damaging or degrading electronic equipment
- By hampering the operator's ability to safely shutdown the plant
- By initiating automatic fire protection systems in areas away from the fire

Reference [23] identifies possible reduction of manual fire-fighting effectiveness and misdirected suppression efforts as the central issue in GSI-148. Manual fire-fighting was not credited in the analysis. Thus, the issue of manual fire-fighting effectiveness is not addressed in this TER.

2.4.3 GSI-156, "Systematic Evaluation Program (SEP)"

Reference [23] provides the description of each SEP issue stated below, and delineates the scope of information that may be reported in an IPEEE submittal relevant to each such issue. The objective of this subsection is only to identify the location in the IPEEE submittal where information having potential relevance to GSI-156 may be found.

Settlement of Foundations and Buried Equipment

Description of the Issue [23]: The objective of this SEP issue is to assure that safety-related structures, systems and components are adequately protected against excessive settlement. The scope of this issue includes review of subsurface materials and foundations, in order to assess the potential static and seismically induced settlement of all safety-related structures and buried equipment. Excessive settlement or collapse of foundations could result in failures of structures, interconnecting piping, or control systems, such that the capability to safely shutdown the plant or mitigate the consequences of an accident could be comprised. This issue, applicable mainly to soil sites, involves two specific concerns:

- potential impact of static settlements of foundations and buried equipment where the soil might not have been properly prepared, and
- seismically induced settlement and potential soil liquefaction following a postulated seismic event.

Since static settlements are not believed to be a concern, the focus of this issue (when considering relevant information in IPEEEs) should be on seismically induced settlements and soil liquefaction. It is anticipated that full-scope seismic IPEEEs will address these concerns, following the guidance in EPRI NP-6041.

Turkey Point is a reduced-scope plant, and safety-related plant structures are founded predominantly on rock. The IPEEE submittal provides no discussion of the potential and effects for seismically induced settlements or soil liquefaction. Information on site geology can be found in Section 2.2 of Reference [5].

Dam Integrity and Site Flooding

Description of the Issue [23]: The objective of this issue is to ensure the ability of a dam to prevent site flooding and to ensure a cooling water supply. The safety functions would normally include remaining stable under all conditions of reservoir operation, controlling seepage to prevent excessive uplifting water pressures or erosion of soil materials, and providing sufficient freeboard and outlet capacity to prevent overtopping. Therefore, the focus is to assure that adequate safety margins are available under all loading conditions, and uncontrolled releases of retained water are prevented. The concern of site flooding resulting from non-seismic failure of an upstream dam (i.e., caused by high winds, flooding, and other events) is addressed as part of the SEP issue "site hydrology and ability to withstand floods." The concerns of site flooding resulting from the seismic failure of an upstream dam and loss of the ultimate heat sink caused by the seismically induced failure of a downstream dam should be addressed in the seismic portion of the IPEEE. The guidance for performing such evaluations is provided in Section 7 of EPRI NP-6041. As requested in NUREG-1407, the licensee's IPEEE submittal should provide specific information addressing this issue, if applicable to its plant. Information included for resolution of USI A-45 is also applicable to this concern.

The Turkey Point IPEEE submittal does not indicate whether or not the failure of any dam would have an impact on plant operations. The submittal provides no information regarding seismically induced dam failures.

Site Hydrology and Ability to Withstand Floods

Description of the Issue [23]: The objective of this issue is to identify the site hydrologic characteristics, in order to ensure the capability of safety-related structures to withstand flooding, to ensure adequate cooling water supply, and to ensure in-service inspection of water-control structures. This issue involves assessing the following:

- Hydrologic conditions - to assure that plant design reflects appropriate hydrologic conditions.
- Flooding potential and protection - to assure that the plant is adequately protected against floods.
- Ultimate heat sink - to assure an appropriate supply of cooling water during normal and emergency shutdown.

As requested in NUREG-1407, the licensee's IPEEE submittal should provide information addressing these concerns. The concern related to in-service inspection of water-control structures, a compliance issue, is not being covered in the IPEEE.

The Turkey Point IPEEE submittal (Section 5.2) has included a discussion of external floods, including effects of storm surge and increased probable maximum precipitation. This discussion references the detailed evaluation of hurricane effects, which is fully documented in Section 3.8 of the June 25, 1991 IPE submittal.

Industrial Hazards

Description of the Issue [23]: The objective of this issue is to ensure that the integrity of safety-related structures, systems, and components would not be jeopardized due to accident hazards from nearby facilities. Such hazards include: shock waves from nearby explosions, releases of hazardous gases, or chemicals resulting in fires or explosions, aircraft impacts, and missiles resulting from nearby explosions. As requested in NUREG-1407, the licensee's IPEEE submittal should provide information addressing this issue.

The Turkey Point IPEEE submittal (Section 5.3) includes the following information of relevance to this issue: Section 5.3.4.1 of the submittal discusses aircraft crashes; Section 5.3.4.2 of the submittal discusses potential explosions; and Section 5.3.4.3 of the submittal discusses potential toxic chemical releases.

Tornado Missiles

Description of the Issue [23]: The objective of this issue is to assure that plants constructed prior to 1972 (SEP plants) are adequately protected against tornadoes. Safety-related structures, systems, and components need to be able to withstand the impact of an appropriate postulated spectrum of tornado-generated missiles. As requested in NUREG-1407, the licensee's IPEEE submittal should provide information addressing this issue.

The Turkey Point IPEEE has involved an evaluation of tornadoes, including tornado-induced missiles. Sections 5.1.2 to 5.1.6 of the submittal provide discussion relevant to tornadoes. Section 5.1.4.2 provides specific information on tornado missile hazard frequencies; Section 5.1.5.1 provides specific information on plant vulnerability to tornado-induced missiles; and Section 5.1.6 provides the specific CDF result for tornado-induced missiles.

Severe Weather Effects on Structures

Description of the Issue [23]: The objective of this issue is to assure that safety-related structures, systems, and components are designed to function under all severe weather conditions to which they may be exposed. Meteorological phenomena to be considered include: straight wind loads, tornadoes, snow and ice loads, and other phenomena judged to be significant for a particular site. As requested in NUREG-1407, the licensee's IPEEE submittal should provide information specifically addressing high winds and floods. Other severe weather conditions (i.e., snow and ice loads) were determined to have insignificant effects on structures (see Chapter 2 of NUREG-1407).

The Turkey Point IPEEE has included evaluations of high winds (straight wind loads, hurricanes, and tornadoes) and external floods. Section 5.1 of the submittal discusses severe winds, hurricanes, and tornadoes, and Section 5.2 of the submittal discusses external floods. In addition, a detailed evaluation of hurricane effects is documented in Section 3.8 of the June 25, 1991 IPE submittal.

Design Codes, Criteria, and Load Combinations

Description of the Issue [23]: The objective of this issue is to assure that structures important to safety should be designed, fabricated, erected, and tested to quality standards commensurate with their safety function. All structures, classified as Seismic Category I, are required to withstand the appropriate design conditions without impairment of structural integrity or the performance of required safety functions. Due to the evolutionary nature of design codes and standards, operating plants may have been designed to codes and criteria which differ from those currently used for evaluating new plants. Therefore, the focus of this issue is to assure that plant Category I structures will withstand the appropriate design conditions (i.e., against seismic, high winds, and floods) without impairment of structural integrity or the performance of required safety function. As part of the IPEEE, licensees are expected to perform analyses to identify potential severe accident vulnerabilities associated with external events (i.e., assess the seismic capacities of their plants either by performing seismic PRAs or SMAs).

The Turkey Point IPEEE has included an evaluation of potential vulnerabilities associated with external events. The submittal does not systematically identify codes, criteria, and load combinations used in design. However, Sections 2.5, and 3.1 to 3.5 of Reference [5] provide some information related to seismic design of structures and equipment; Section 5.1.5 of the IPEEE submittal provides some information related to wind design of structures; Section 3.8 of the June 25, 1991 IPE submittal contains some information related to design conditions for withstanding floods; and Section 5.3 of the submittal provides some limited information on design criteria related to transportation and nearby facility accidents, including explosions.

Seismic Design of Structures, Systems, and Components

Description of the Issue [23]: The objective of this SEP issue is to review and evaluate the original seismic design of safety-related structures, systems, and components, to ensure the capability of the plant to withstand the effects of a Safe Shutdown Earthquake (SSE).

The Turkey Point IPEEE is based on the seismic adequacy evaluation performed as part of the licensee's resolution of USI A-46 concerns (Reference [5]). Sections 2.5 and 3 of Reference [5] provide some information related to the seismic design of structures and components, and Section 4 of Reference [5] provides a description of the approach and findings of the seismic adequacy evaluation.

Shutdown Systems and Electrical Instrumentation and Control Features

Description of the Issue [23]: The issue on shutdown systems is to address the capacity of plants to ensure reliable shutdown using safety-grade equipment. The issue on electrical instrumentation and control is to assess the functional capabilities of electrical instrumentation and control features of systems required for safe shutdown, including support systems. These systems should be designed, fabricated, installed, and tested to quality standards, and remain functional following external events. In IPEEEs, licensees were requested to address USI A-45, "Shutdown Decay Heat Removal (DHR) Requirements," and to identify potential vulnerabilities associated with DHR systems following the occurrence of external events. The resolution of USI A-45 should address these two issues.

Turkey Point Nuclear Plant had been used as a case study plant by Sandia National Laboratories for probabilistic evaluation of decay heat removal adequacy, in the context of USI A-45. This issue was addressed as part of the IPE and IPEEE, in general, and pertinent information is provided in Section 3.9.3.3 of the June 25, 1991 IPE submittal. Sections 2.1.13 and 2.2.13 of this TER summarize review findings related to USI A-45, respectively, for seismic events and fire events

2.4.4 GSI-172, "Multiple System Responses Program (MSRP)"

Reference [23] provides the description of each MSRP issue stated below, and delineates the scope of information that may be reported in an IPEEE submittal relevant to each such issue. The objective of this subsection is only to identify the location in the IPEEE submittal where information having potential relevance to GSI-172 may be found.

Common Cause Failures (CCFs) Related to Human Errors

Description of the Issue [23]: CCFs resulting from human errors include operator acts of commission or omission that could be initiating events, or could affect redundant safety-related trains needed to mitigate the events. Other human errors that could initiate CCFs include: manufacturing errors in components that affect redundant trains; and installation, maintenance or testing errors that are repeated on redundant trains. In IPEEEs, licensees were requested to address only the human errors involving operator recovery actions following the occurrence of external initiating events.

A very limited discussion of operator recovery actions, following a seismic event, is provided in Section 4.4 of Reference [5]. Section 3.7 of the June 25, 1995 IPE submittal provides some discussion on the treatment of human recovery actions in the internal fire analysis.

Non-Safety-Related Control System/Safety-Related Protection System Dependencies

Description of the Issue [23]: Multiple failures in non-safety-related control systems may have an adverse impact on safety-related protection systems, as a result of potential unrecognized dependencies between control and protection systems. The concern is that plant-specific implementation of the regulations regarding separation and independence of control and protection systems may be inadequate. The licensees' IPE process should provide a framework for systematic evaluation of interdependence between safety-related and non-safety-related systems, and should identify potential sources of vulnerabilities. The dependencies between safety-related and non-safety-related systems resulting from external events -- i.e., concerns related to spatial and functional interactions -- are addressed as part of "fire-induced alternate shutdown and control room panel interactions," GSI-147, for fire events, and "seismically induced spatial and functional interactions" for seismic events.

Information provided in the Turkey Point IPEEE submittal pertaining to seismically induced spatial and functional interactions is identified below (under the heading *Seismically Induced Spatial and Functional Interactions*), whereas information pertaining to fire-induced alternate shutdown and control panel interactions has already been identified in Section 2.4.1 of this TER.

Heat/Smoke/Water Propagation Effects from Fires

Description of the Issue [23]: Fire can damage one train of equipment in one fire zone, while a redundant train could potentially be damaged in one of following ways:

- Heat, smoke, and water may propagate (e.g., through HVAC ducts or electrical conduit) into a second fire zone, and damage a redundant train of equipment.
- A random failure, not related to the fire, could damage a redundant train.
- Multiple non-safety-related control systems could be damaged by the fire, and their failures could affect safety-related protection equipment for a redundant train in a second zone.

A fire can cause unintended operation of equipment due to hot shorts, open circuits, and shorts to ground. Consequently, components could be energized or de-energized, valves could fail open or closed, pumps could continue to run or fail to run, and electrical breakers could fail open or closed. The concern of water propagation effects resulting from fire is partially addressed in GI-57, "Effects of Fire Protection System Actuation on Safety-Related Equipment." The concern of smoke propagation effects is addressed in GSI-148. The concern of alternate shutdown/control room interactions (i.e., hot shorts and other items just mentioned) is addressed in GSI-147.

Information provided in the Turkey Point IPEEE submittal pertaining to GSI-147 and GSI-148 has already been identified in Sections 2.4.1 and 2.4.2 of this TER. Section 3.7.4 of the June 25, 1991 IPE submittal presents some limited information pertinent to this issue.

Effects of Fire Suppression System Actuation on Non-Safety-Related and Safety-Related Equipment

Description of the Issue [23]: Fire suppression system actuation events can have an adverse effect on safety-related components, either through direct contact with suppression agents or through indirect interaction with non-safety related components.

Item 2 of Section 3.7.4 of the June 25, 1991 IPE submittal [11] presents some very limited information pertinent to this issue.

Effects of Flooding and/or Moisture Intrusion on Non-Safety-Related and Safety-Related Equipment

Description of the Issue [23]: Flooding and water intrusion events can affect safety-related equipment either directly or indirectly through flooding or moisture intrusion of multiple trains of non-safety-related equipment. This type of event can result from external flooding events, tank and pipe ruptures, actuations of fire suppression systems, or backflow through parts of the plant drainage system. The IPE process addresses the concerns of moisture intrusion and internal flooding (i.e., tank and pipe ruptures or backflow through part of the plant drainage system). The guidance for addressing the concern of external flooding is provided in Chapter 5 of NUREG-1407, and the concern of actuations of fire suppression systems is provided in Chapter 4 of NUREG-1407.

The following information is provided relevant to this issue: the Turkey Point IPEEE submittal discusses external floods in Section 5.2; Section 3.8 of the IPE [11] presents information on flooding due to hurricanes; and Item 2 of Section 3.7.4 of the IPE presents some very limited information regarding seismically induced inadvertent actuation of fire suppression systems.

Seismically Induced Spatial and Functional Interactions

Description of the Issue [23]: Seismic events have the potential to cause multiple failures of safety-related systems through spatial and functional interactions. Some particular sources of concern include: ruptures in small piping that may disable essential plant shutdown systems; direct impact of non-seismically qualified structures, systems, and components that may cause small piping failures; seismic functional interactions of control and safety-related protection systems via multiple non-safety-related control systems' failures; and indirect impacts, such as dust generation, disabling essential plant shutdown systems. As part of the IPEEE, it was specifically requested that seismically induced spatial interactions be addressed during plant walkdowns. The guidance for performing such walkdowns can be found in EPRI NP-6041.

The Turkey Point seismic adequacy evaluation (Reference [5]) has included a seismic walkdown which investigated the potential for adverse physical interactions. Relevant information can be found in Section 4.7 (particularly Section 4.7.2.3) of Reference [5].

Seismically Induced Fires

Description of the Issue [23]: Seismically induced fires may cause multiple failures of safety-related systems. The occurrence of a seismic event could create fires in multiple locations, simultaneously degrade fire suppression capability, and prevent mitigation of fire damage to multiple safety-related systems. Seismically induced fires is one aspect of seismic-fire interaction concerns, which is addressed as part of the Fire Risk Scoping Study (FRSS) issues. (IPEEE guidance specifically requested licensees

to evaluate FRSS issues.) In IPEEEs, seismically induced fires should be addressed by means of a focused seismic-fire interactions walkdown that follows the guidance of EPRI NP-6041.

A very brief discussion of seismic-fire interactions, including mention of seismically induced fires, is provided in Section 3.7.4 of the June 25, 1991 IPE submittal [11]; however, no evaluation of seismically induced fires is provided as part of the Turkey Point IPEEE submittal.

Seismically Induced Fire Suppression System Actuation

Description of the Issue [23]: Seismic events can potentially cause multiple fire suppression system actuations which, in turn, may cause failures of redundant trains of safety-related systems. Analyses currently required by fire protection regulations generally only examine inadvertent actuations of fire suppression systems as single, independent events, whereas a seismic event could cause multiple actuations of fire suppression systems in various areas.

Item 2 of Section 3.7.4 of the June 25, 1991 IPE submittal presents some very limited information regarding seismically induced actuation of fire suppression systems.

Seismically Induced Flooding

Description of the Issue [23]: Seismically induced flooding events can potentially cause multiple failures of safety-related systems. Rupture of small piping could provide flood sources that could potentially affect multiple safety-related components simultaneously. Similarly, non-seismically qualified tanks are a potential flood source of concern. IPEEE guidance specifically requested licensees to address this issue.

The Turkey Point IPEEE submittal has not included a discussion of seismically induced flooding.

Seismically Induced Relay Chatter

Description of the Issue [23]: Essential relays must operate during and after an earthquake, and must meet one of the following conditions:

- remain functional (i.e., without occurrence of contact chattering);
- be seismically qualified; or
- be chatter acceptable.

It is possible that contact chatter of relays not required to operate during seismic events may produce some unanalyzed faulting mode that may affect the operability of equipment required to mitigate the event. IPEEE guidance specifically requested licensees to address the issue of relay chatter.

As noted in Section 2.1.9 of this TER, the Turkey Point IPEEE submittal does not mention relay chatter evaluation. However, during NRC's USI A-46 review, it was revealed that the licensee had assessed bad actor relays, verified mountings of relays, and demonstrated that there were no deleterious effects of chatter of bad actor relays. The NRC accepted the licensee's relay evaluation for USI A-46 resolution.

Evaluation of Earthquake Magnitudes Greater than the Safe Shutdown Earthquake

Description of the Issue [23]: The concern of this issue is that adequate margin may not have been included in the design of some safety-related equipment. As part of the IPEEE, all licensees are expected to identify potential seismic vulnerabilities or assess the seismic capacities of their plants either by performing seismic PRAs or seismic margins assessments (SMAs). The licensee's evaluation for potential vulnerabilities (or unusually low plant seismic capacity) due to seismic events should address this issue.

Turkey Point is designated as a reduced-scope plant in NUREG-1407, and consistent with the relevant guidelines for a reduced-scope plant, the IPEEE has considered seismic input equivalent to the SSE level. Earthquake loads in excess of the SSE have not been considered.

Effects of Hydrogen Line Ruptures

Description of the Issue [23]: Hydrogen is used in electrical generators at nuclear plants to reduce windage losses, and as a heat transfer agent. It is also used in some tanks (e.g., volume control tanks) as a cover gas. Leaks or breaks in hydrogen supply piping could result in the accumulation of a combustible mixture of air and hydrogen in vital areas, resulting in a fire and/or an explosion that could damage vital safety-related systems in the plants. It should be anticipated that the licensee will treat the hydrogen lines and tanks as potential fixed fire sources as described in EPRI's FIVE guide, assess the effects of hydrogen line and tank ruptures, and report the results in the fire portion of the IPEEE submittal.

Section 5.3.4.2 of the Turkey Point IPEEE submittal evaluates a hydrogen trailer as a potential explosion sources. No discussion pertaining to hydrogen line ruptures is provided in the submittal.

3 OVERALL EVALUATION AND CONCLUSIONS

3.1 Seismic

The approach chosen by the licensee for responding to the seismic IPEEE does not address all relevant issues and concerns for Turkey Point Nuclear Plant, a reduced-scope site. A comparison of major features of the FPL seismic adequacy program with the guidelines for a reduced-scope seismic evaluation, is summarized in Table 3.1 below. As can be seen from this table, the primary deficiencies of the FPL approach are: a significantly lesser scope of components in the FPL approach; a limited treatment of human actions in the FPL study; and no treatment of containment systems in the FPL program.

Table 3.1 Comparison of FPL's Site-Specific Seismic IPEEE Program Versus NUREG-1407 Recommended Guidelines for a Reduced-Scope Seismic Evaluation

Element of IPEEE Evaluation	Reduced-Scope Evaluation Guidelines	FPL's Site-Specific Seismic Adequacy Program
Walkdown	Scope should include all SSEL active components and passive components (structures, raceways, heat exchangers, tanks, piping, etc.) needed to ensure complete preferred and alternate success paths	Scope includes SSEL active components and some passive components (tanks, heat exchangers); component list appears incomplete; selected success paths not identified. USI A-46 treatment of electrical raceways approved by the NRC [7].
Relay Evaluation	USI A-46 evaluation for USI A-46 plant; No evaluation for non-USI A-46 plant	Bad actor evaluation for Turkey Point, Units 3 and 4, approved by NRC for USI A-46 [7].
Soil Failures	No evaluation is necessary	No evaluation (rock site)
Screening Criteria	SRT judgment; GIP screening guidance; Anchorage check based on SSE spectrum and FSAR in-structure response spectrum (IRS)	SRT judgment; SSRAP bounding spectrum; Anchorage check based on SSE spectrum and FSAR IRS
Seismic Input	SSE spectrum and FSAR IRS (or new mean plus one-sigma IRS)	SSE spectrum and FSAR IRS
Evaluation of Outliers	GIP provisions for USI A-46 Items; FSAR requirements for non-USI A-46 items	Conservative calculation of capacity versus demand; demand based on conservative use of SSE spectrum and FSAR IRS; HCLPF calculations for large flat-bottomed tanks at Turkey Point Nuclear Plant
Non-Seismic Failures and Human Actions	These should be qualitatively addressed; success paths are chosen to screen out vulnerability to these items	Limited qualitative evaluation of actions associated with success path
Containment Performance Assessment	Walkdown, screening, and outlier evaluation of containment structure and components of containment systems	No evaluation
USI A-45	Walkdown, screening, and evaluation of decay heat removal outliers	No specific evaluation; only partially addressed in chosen success path
GI-131	Walkdown, screening, and evaluation of seismic adequacy of flux mapping system	Addressed by previous upgrade

In addition, the format for documenting the seismic IPEEE was not well structured, and did not follow the recommendations of NUREG-1407.

Despite these significant deficiencies, the Turkey Point seismic evaluation does, nonetheless, address some meaningful IPEEE-related concerns, and has resulted in a number of plant seismic safety enhancements. Furthermore, the NRC has already approved many aspects of the licensee's seismic adequacy evaluation approach for USI A-46 resolution [6,7] that pertain also to the seismic IPEEE. Based on this submittal-only review, and in consideration of the NRC's findings for USI A-46, the following items are identified as the primary strengths and weaknesses of the seismic IPEEE submittal for Turkey Point Nuclear Plant:

<u>Strengths</u>

1. The study implements a meaningful approach for screening and outlier evaluation of the limited set of components it addresses.

2. The use of highly experienced seismic walkdown experts has been consistent with the study's heavy reliance on SRT judgments.

3. A number of outliers have been identified, and meaningful corrective safety enhancements have been proposed.

<u>Weaknesses</u>

1. The SSEL is deficient.

2. A seismic containment performance assessment was not conducted.

3. The treatment of human actions is deficient.

4. The submittal does not provide adequate documentation of seismic-fire/flood interaction concerns, including component-specific walkdown findings.

5. The seismic IPEEE is incomplete with respect to reduced-scope evaluation recommendations found in NUREG-1407.

6. The seismic IPEEE submittal is not documented in accordance with the format recommended in NUREG-1407, Appendix C.

3.2 <u>Fire</u>

The licensee has expended a considerable effort in its preparation of the fire analysis portion of the IPEEE. The licensee has employed an appropriate methodology for conducting the fire analysis; the EPRI FIVE methodology has been used for this purpose.

The following are the strengths and weaknesses of the submittal:

<u>Strengths</u>

1. The overall presentation is clear and well-organized. Tables and figures provide partial support of the analysis and the conclusions.

2. An acceptable methodology has been used. Even though the 1990 version of FIVE was modified per NRC comments, in the reviewer's judgement the results of the fire analysis would be minimally affected, if the most recent version of FIVE were to be employed for the analysis.

Weaknesses

1. The failure probability for the fire suppression system may not have been used properly. If a critical set of redundant cables and equipment are grouped together in a compartment, the successful operation of the fire suppression system may not matter.

2. Probability of failure of redundant equipment, and models used for arriving at the conditional probability of core damage given a fire scenario, have not been explained in sufficient detail.

3. Active fire barriers were assumed to be highly reliable.

4. The fire occurrence frequency for some of the fire zones may be too small, thus leading to premature screening.

5. Local manual actions needed to recover from fire-induced failures have not been analyzed using probabilistic methods.

Certainly, notwithstanding the above observations, the licensee has gained valuable experience from the effort of inspecting the entire plant, with the exception of the containment, for potential fire vulnerabilities. As a final observation, the resolution of the Thermo-Lag issue may have a profound impact on the results of the IPEEE fire analysis.

3.3 HFO Events

The HFO submittal has generally followed recommended IPEEE submittal guidelines regarding basic steps of analyzing and reporting potential accident scenarios. Several specific weaknesses, however, have been identified by this review. These are summarized below:

1. It is unclear whether or not the missile simulation code, TORMIS, used in the IPE for the hurricane analysis (and which was referenced in the submittal), was re-executed for wind speeds greater than 200 mph when the tornado analysis was performed.

2. A summary of the walkdown procedures used by the licensee, and the qualification of the team members performing the walkdown, are not provided in the submittal.

3. Documentation is lacking as to whether or not the Air Force base was contacted to ascertain if there are additional explosion hazards present at the facility, other than jet fuel.

4. It appears that the amount of time that elapses before a natural gas leak is discovered may not have been factored into the estimate of the amount of time with which the Florida Gas Transmission Company can isolate a natural gas pipe rupture.

5. There appears to be no basis for the hydrogen explosion initiating event frequency.

6. The submittal does not provide a comprehensive list of plant changes made since the issuance of the operating license, that would be relevant to each of the analyses reported in the submittal.

4 IPEEE INSIGHTS, IMPROVEMENTS, AND COMMITMENTS

4.1 Seismic

The key seismic IPEEE findings are primarily walkdown related; few quantitative insights have been derived from the seismic evaluations. Thus, no values for seismic core damage frequency, plant-level fragility capacity nor plant-level HCLPF capacity have been estimated as a result of the seismic IPEEE.

The seismic adequacy evaluation for Turkey Point, Units 3 and 4, revealed a number of outliers for which safety enhancements have been proposed or implemented in response to USI A-46. Enhancements for IPEEE-only components (i.e., components outside the scope of USI A-46, but within the scope of IPEEE) were not addressed. In addition, containment performance evaluation and evaluation of human actions were not included as part of the licensee's treatment of seismic IPEEE concerns.

Table 4.1 below (reproduced from Table 5.0 of the Turkey Point Nuclear Plant seismic adequacy evaluation submittal) summarizes the 35 components identified as outliers, and describes the pertaining actions that were taken. The table indicates that 26 anchorage/support concerns, 12 interaction hazards, and 2 functional concerns were identified. In addition to these items, some cases of poor seismic "housekeeping" were observed during seismic walkdowns, and these were documented by the seismic review team.

As a result of an NRC site audit conducted during the USI A-46 review process, the licensee has also agreed to address additional issues related to: an anchorage concern, corrosion concerns requiring maintenance, a concern with interaction of station batteries, and the need for a strict housekeeping program.

The licensee's submittal reports the results of HCLPF calculations that were performed for a number of large storage tanks; these calculations initially produced the following results:

- Diesel Oil Storage Tank; HCLPF=0.21g (after upgrade)
- Condensate Storage Tanks; HCLPF=0.11g (after upgrade)
- Refueling Water Storage Tanks; HCLPF=0.11g (after upgrade)

Hence, even after implementation of safety upgrades, the reported HCLPF capacities for the CSTs and RWSTs were found to fall below the level of the design-basis earthquake for Turkey Point Nuclear Plant. In response to the NRC's USI A-46 review process, the licensee re-evaluated the capacities of CSTs and RWSTs, and reported them to meet the seismic design basis of 0.15g PGA. In its review of the revised calculations, the NRC determined that the tank HCLPF capacities exceeded 0.12g PGA and were sufficiently close to the plant's seismic design level. The licensee has not reported HCLPF capacities for other outlier components being upgraded.

Regarding GI-131 ("Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants"), FPL has stated that lateral restraint was added to the movable support assembly of the flux mapping system in 1989. The licensee considers this issue to be closed. USI A-45

("Shutdown Decay Heat Removal Requirements") is also applicable to Turkey Point Nuclear Plant, but was not addressed directly in the licensee's IPEEE submittal report.

Table 4.1 Seismic Outlier Issues and Proposed/Completed Resolutions at Turkey Point Units 3 and 4

EQUIPMENT DESCRIPTION	OUTLIER ISSUES	SRT RECOMMENDED RESOLUTION	ACTION TAKEN
Intake Cooling Water Pump Item No. 6; I.D. No. 3B	1) Pump shaft length longer than can be screened by SSRAP report. 2) Cast iron fittings on pump. 3) Anchorage needs verification. 4) Interaction with fossil unit stack may cause failure.	1) Evaluate shaft for adequate length and clearances. 2) Check stresses on fittings from loads of attached piping. 3) Verify anchorage with calculation. 4) Check adequacy of fossil stack.	1) TBA
Intake Cooling Water Pump Item No. 7; I.D. No. 4B	1) Pump shaft length longer than can be screened by SSRAP report. 2) Cast iron fittings on pump. 3) Anchorage needs verification. 4) Interaction with fossil unit stack may cause failure.	1) Evaluate shaft for adequate length and clearances. 2) Check stresses on fittings from loads of attached piping. 3) Verify anchorage with calculation. 4) Check adequacy of fossil stack.	1) TBA
Diesel Oil Storage Tank Item No. 10; I.D. No. N/A	1) Anchorage adequacy. 2) Interaction with fossil unit stack may cause failure.	1) Replace chair plates with 1-1/4" thick plates, and evaluate further. 2) Check adequacy of fossil stack.	1) Chair plates upgraded per PC/M 91-169 2) Fossil stack adequate as per FPL safety evaluation, TBA.
Boric Acid Storage Tank Item No. 11; I.D. No. B	1) Platform adequacy for torsional loads.	1) Check platform adequacy for torsion, and upgrade if required.	1) Platform upgraded as per PC/M 90-440 and PC/M 90-441.
Condensate Storage Tank (Unit 3) Item No. 12; I.D. No. 3	1) Anchorage adequacy.	1) Replace chair plates with 1-1/4" thick plates, and evaluate further.	1) Chair plates upgraded as per PC/M 91-170.
Condensate Storage Tank (Unit 4) Item No. 13; I.D. No. 4	1) Anchorage adequacy.	1) Replace chair plates with 1-1/4" thick plates, and evaluate further.	1) Chair plates upgraded as per PC/M 91-171.
Refueling Water Storage Tank (Unit 3) Item No. 14; I.D. No. 3	1) Anchorage adequacy.	1) Replace chair plates with 1-1/4" thick plates, and evaluate further.	1) Chair plates upgraded as per PC/M 91-172.
Refueling Water Storage Tank (Unit 4) Item No. 15; I.D. No. 4	1) Anchorage adequacy.	1) Replace chair plates with 1-1/4" thick plates, and evaluate further.	1) Chair plates upgraded as per PC/M 91-173.
Emergency Diesel Generator Day Tank Item No. 16; I.D. No. B	1) Glass site tube interaction.	1) Replace glass sight tube with non-breakable material.	1) TBA
Component Cooling Water Surge Tank (Unit 3) Item No. 17; I.D. No. 3	1) Platform adequacy.	1) Check platform adequacy, and upgrade if required.	1) Platform to be upgraded as per PC/M 90-471.
Component Cooling Water Surge Tank (Unit 4) Item No. 18; I.D. No. 4	1) Platform adequacy.	1) Check platform adequacy, and upgrade if required.	1) Platform to be upgraded as per PC/M 90-472.

Table 4.1 (continued) Seismic Outlier Issues and Proposed/Completed Resolutions at Turkey Point Units 3 and 4

EQUIPMENT DESCRIPTION	OUTLIER ISSUES	SRT RECOMMENDED RESOLUTION	ACTION TAKEN
Emergency Diesel Generator Skid Item No. 19; I.D. No. B	1) Glass sight tube interaction	1) Replace glass sight tube with non-breakable material.	1) TBA
Emergency Diesel Generator Air Start Tanks Item No. 20; I.D. No. B	1) Seismic interaction - threaded pipe for air supply is not rigidly supported.	1) Complete plant work order (PWO) already written for the support.	1) Air supply and supports replaced as per PC/M 86-155 and PC/M 86-190.
480V Motor Control Center 3B Item No. 23; I.D. No. 3B06	1) Seal welded anchorage; inadequate in tension.	1) Upgrade anchorage.	1) Anchorage upgraded as per PC/M 91-178.
480V Motor Control Center 4B Item No. 24; I.D. No. 4B06	1) No anchorage.	1) Add anchorage	1) Anchorage upgraded as per PC/M 91-179.
480V Motor Control Center D Item No. 25; I.D. No. B08	1) Inadequate anchorage for overturning.	1) Brace top of MCC to concrete wall.	1) Anchorage upgraded as per PC/M 91-178.
4.16kV Switchgear 3B Item No. 26; I.D. No. 3AB	1) No anchorage.	1) Add anchorage.	1) Anchorage upgraded as per PC/M 91-174.
4.16kV Switchgear 4B Item No. 27; I.D. No. 4AB	1) No anchorage.	1) Add anchorage.	1) Anchorage upgraded as per PC/M 91-175.
480V HVPDS Load Center 3B (Includes Transformer) Item No. 28; I.D. No. 3B02	1) Cannot determine anchorage.	1) Add anchorage.	1) New load center installed as per PC/M 89-532, and new anchorage installed as per PC/M 91-176.
480V HVPDS Load Center 3D (Includes Transformer) Item No. 29; I.D. No. 3B04	1) Cannot determine anchorage	1) Verify anchorage, and upgrade if required	1) New load center installed as per PC/M 89-532, and new anchorage installed as per PC/M 91-176.
480V HVPDS Load Center 4B (Includes Transformer) Item No. 30; I.D. No. 4B02	1) No anchorage.	1) Add anchorage.	1) New load center installed as per PC/M 89-533, and new anchorage installed as per PC/M 91-177.
480V HVPDS Load Center 4D (Includes Transformer) Item No. 31; I.D. No. 4B04	1) No anchorage.	1) Add anchorage.	1) New load center installed as per PC/M 89-533, and new anchorage installed as per PC/M 91-177.

Table 4.1 (continued) Seismic Outlier Issues and Proposed/Completed Resolutions at Turkey Point Units 3 and 4

EQUIPMENT DESCRIPTION	OUTLIER ISSUES	SRT RECOMMENDED RESOLUTION	ACTION TAKEN
Battery Rack 3A Item No. 37; I.D. No. 3D03	1) No spacers on east end of battery rack. 2) Shade on lights may fall on batteries. 3) Block walls not evaluated by SRT.	1) Add spacers on east end of battery rack. 2) Add tie wire to lights. 3) Verify that block walls are included in FPL's IE 80-11 program.	1) Spacers added as per FPL letters JPN-PTN-92-5261 and 5707. 2) Tie wires added as per PC/M 91-182. 3) FPL verified that block walls are included in IE 80-11 program; Block Walls C30-1, C30-2, and C30-4.
Battery Rack 3B Item No. 38; I.D. No. 3D24	1) No spacers on east end of battery rack. 2) Shade on lights may fall on batteries. 3) Block walls not evaluated by SRT.	1) Add spacers on east end of battery rack. 2) Add tie wire to lights. 3) Verify that block walls are included in FPL's IE 80-11 program.	1) Spacers added as per FPL letters JPN-PTN-92-5261 and 5707. 2) Tie wires added as per PC/M 91-182. 3) FPL verified that block walls are included in IE 80-11 program; Block Walls A42-2, C42-16, C42-17, and C42-18.
Battery Rack 4B Item No. 39; I.D. No. 4D03	1) No spacers on east end of battery rack. 2) Shade on lights may fall on batteries. 3) Block walls not evaluated by SRT.	1) Add spacers on east end of battery rack. 2) Add tie wire to lights. 3) Verify that block walls are included in FPL's IE 80-11 program.	1) Spacers added as per FPL letters JPN-PTN-92-5261 and 5707. 2) Tie wires added as per PC/M 91-183. 3) FPL verified that block walls are included in IE 80-11 program; Block Walls C30-2, C30-3, and C30-4.
Battery Rack 4A Item No. 40; I.D. No. 4D24	1) No spacers on east end of battery rack. 2) Shade on lights may fall on batteries. 3) Block walls not evaluated by SRT.	1) Add spacers on east end of battery rack. 2) Add tie wire to lights. 3) Verify that block walls are included in FPL's IE 80-11 program.	1) Spacers added as per FPL letters JPN-PTN-92-5261 and 5707. 2) Tie wires added as per PC/M 91-183. 3) FPL verified that block walls are included in IE 80-11 program; Block Walls A42-2, C42-15, C42-16, and C42-18.
Distribution Panels / Bus Item No. 41; I.D. No. 3D01	1) One loose anchor bolt.	1) Tighten loose bolt.	1) Bolt disposition as per PWO 93-010843.
Distribution Panels / Bus Item No. 43; I.D. No. 4D01	1) Three loose anchor bolts.	1) Tighten loose bolts.	1) Bolt disposition as per PWO 93-010844.

Table 4.1 (continued) Seismic Outlier Issues and Proposed/Completed Resolutions at Turkey Point Units 3 and 4

EQUIPMENT DESCRIPTION	OUTLIER ISSUES	SRT RECOMMENDED RESOLUTION	ACTION TAKEN
Sequencer 3B Item No. 50; I.D. No. 3C23B	1) Additional top brackets as found for Sequencer 3A (Item 49) would provide added assurance and strength. (This item had only one bracket.)	1) Add top bracket as found for Sequencer 3A (Item 49).	1) Bracket added as per PC/M 91-180.
Sequencer 4A Item No. 51; I.D. No. 4C23A	1) Additional top brackets as found for Sequencer 3A (Item 49) would provide added assurance and strength. (This item had no brackets.)	1) Add two top brackets as found for Sequencer 3A (Item 49).	1) Bracket added as per PC/M 91-181.
Sequencer 4B Item No. 52; I.D. No. 4C23B	1) Additional top brackets as found for Sequencer 3A (Item 49) would provide added assurance and strength. (This item had no brackets.)	1) Add two top brackets as found for Sequencer 3A (Item 49).	1) Bracket added as per PC/M 91-181.
Component Cooling Water Heat Exchanger Item No. 53; I.D. No. 3B	1) SRT could not verify reinforcement steel design of pedestal.	1) Verify adequacy of pedestal design.	1) FPL verified pedestal adequacy by calculations C-SJ511-01 and 02.
Component Cooling Water Heat Exchanger Item No. 54; I.D. No. 4B	1) SRT could not verify reinforcement steel design of pedestal.	1) Verify adequacy of pedestal design.	1) FPL verified pedestal adequacy by similarity with Item 53.
Vertical Panel 3B Item No. 55; I.D. No. 3C05, 3C06	1) Interaction hazard; metal egg crate ceiling may fall on operators.	1) Clip in metal egg crate sections of ceiling.	1) TBA
Vertical Panel 4B Item No. 56; I.D. No. 4C05, 4C06	1) Interaction hazard; metal egg crate ceiling may fall on operators.	1) Clip in metal egg crate sections of ceiling.	1) TBA

4.2 Fire

Overall, the licensee has concluded that there are no significant fire vulnerabilities at Turkey Point Nuclear Plant, Units 3 and 4. With the exception of the control room, cable spreading room, control rod equipment rooms, and ICW intake structures, all fire zones and areas were screened out based on either a 10^{-6}/ry bound on core damage frequency, or on a lack of safe shutdown equipment for the zone/area.

The control room and cable spreading room contain a large array of equipment and cables that are needed for safe shutdown of the plant. For the core damage frequency associated with fires in the control rooms and cable spreading room, the licensee cites several conservative assumptions that were made in the analysis, pertaining to assignments of fire occurrence rate and fire severity, as well as to the availability of alternate shutdown panels. The licensee concludes that these two areas (i.e., control room and cable spreading room) do not pose a fire vulnerability.

The reactor control rod equipment rooms contain the control cables for those valves that are needed for RCP seal integrity. A qualitative analysis was presented for this room. It is claimed that the operators will take control of CCW and the charging pump RCP seal injection, and manually open (via hand-wheels) the proper valves.

At the intake structures, a fire may fail the ICW system and lead to core damage, if recovery actions are not undertaken in a timely fashion. Fire propagation modeling has been performed for fire events at the intake cooling water structure. In the case of loss of the ICW system, RCP seal failure can be prevented by either re-activating the CCW, by using the cross-tie with the other unit; or by re-activating the "B" charging pump, by making a hose connection between the pump oil cooler and the service water system.

The entire fire analysis effort, of course, has provided an excellent opportunity for licensee engineers to better understand the characteristics of the plant, how the plant would behave under fire conditions, and what human actions would be necessary to protect the core from any adverse effects.

4.3 HFO Events

It is stated by the licensee that no HFO vulnerabilities exist at Turkey Point Nuclear Plant. The dominant HFO contributor to risk, storm surge, has been addressed, and several modifications made. Primarily, the "Natural Emergencies" Emergency Plant Implementation Procedure (EPIP) No. 20106 was enhanced. Within this procedure, additional guidance was provided to cope with the effects of severe storms. This procedure was in place, and was cited by the licensee, as contributing significantly to the preparation and mitigation of the effects of Hurricane Andrew. This procedure was not provided in the IPEEE submittal for review.

In 1991, after discovery of the issue of storm surge, the existing flood walls and stop logs at Turkey Point were refurbished. In addition, the Unit 3 emergency diesel generator (EDG) fuel oil transfer pump was raised to an elevation that reduces its vulnerability to storm surge.

As further consequence of the high winds analysis, and as a result of Hurricane Andrew, the Units 1 and 2 (fossil plant) stacks were reinforced to a design wind load of 225 mph. Other modifications and procedural enhancements were performed as a result of Hurricane Andrew, but the submittal does not specify these modifications.

5 IPEEE EVALUATION AND DATA SUMMARY SHEETS

Completed data entry sheets for the Turkey Point Nuclear Plant IPEEE are provided in Tables 5.1 to 5.6. These tables have been completed in accordance with the descriptions in Reference [10]. Table 5.1 lists the overall external events results. Table 5.2 summarizes general seismic data pertaining to the evaluation. Table 5.3 provides the Seismic Success Paths Overview Table, and Table 5.4 summarizes sequence information for PWR Seismic Success Paths. Accident sequence tables for fire events are presented in Tables 5.5 and 5.6. Such tables are not provided for HFO events, since sufficient PRA information was not presented in the IPEEE submittal for these initiators.

Table 5.1
External Events Results

Plant Name: Turkey Point Units 3 and 4

Event	Screening	CDF	Plant HCLPF(g)	Notes
External Fire				
External Flooding	SO			
Extreme Winds	S/SO	10^{-6} to 10^{-4}/ry (Units 3 and 4) $4.81x10^{-7}$/ry (Unit 3); $6.85x10^{-7}$/ry (Unit 4)		Hurricanes (Storm Surge) Tornadoes
Internal Fire	SO			
Nearby Facility Accidents	SO			
Seismic Activity	S			
Transportation Accidents	SO			
Others*	SO			* Lightning

Screening: S = Plant specific analysis; O = Screened out; SO = Bounding analysis

Table 5.2
SSM Seismic Fragility

Plant Name: Turkey Point Units 3 and 4

Review Level Earthquake (g): Reduced-Scope Plant

Spectral Shape: SSE; Housner Spectral Shape (rock); 0.15 g PGA; Vertical = 2/3 (horizontal)
(NUREG-0098, NRC Guide 1.60, 10,000 year LLNL median UHS, Site Specific, or other)

List components and equipments which do not meet RLE (all components) or with lowest HCLPF (less than 10):

Component[1]	HCLPF (g)[2]	Seismic Sequence Description	Seismic Success Path Description
Condensate Storage Tanks (CSTs)	>0.12[3]		
Refueling Water Storage Tanks (RWSTs)	>0.12[3]		
Diesel Fuel Oil Storage Tank	0.21		

1 Not all lowest HCLPFs were reported; reduced scope evaluation.

2 HCLPF results apply to the proposed upgraded plant condition.

3 See NRC's SSER [7]

Table 5.3
PWR Success Path Overview Table

Plant Name: Turkey Point, Units 3 and 4 — 1 Sheet of 1

#	Sequence	PDS	HCLPF (g)*	Init Event	Success Supports	Non-Failed Functions	Attributes
1	Success Path	—	—	T-LOOP	EAC, CCW, ESW		

Init Event (initiator): One of the following: S1, S2, S3, A, V (-xx), T-LOOP, T-RX, T-TT, T-ATWS, T-UHS, T-RCP, T-LNMU, T-LMFW, T-EXFW, T-SLBOC, T-SLBIC, T-SGTR, T-SORV/IORV, T-SSI, T-(Other), or T-(Support System)
(-xx) refers to optional supplementary material.

Success Supports: At most two of the following: AC, ACBU1, ACBU2, ACBU3, AUXC2, AUXC3, AUXC4, CCW, DC, EAC, EDC, ESAS1, ESAS2, ESW, HVAC1, HVAC2, HVAC3, IA, NIT, OA3, OA4, SA, STM, SW2, SW3, SW4, VAC (Field may be blank).

Non-Failed Functions: At most three of the following: SINT, SDEP, SSMU, RCS-BOR, RCS-INT, RCS-DEP, HPI, HPR, LPI, LPR, CPSI, CPSR, CIF, VENT (If a 4th and/or 5th are necessary, use the "Notes" field)

Attributes: At most three of the following: ATWS, BYPASS, TIL, IND-SGTR, SBO, OR HUM (Field may be blank)

* Reduced-Scope Plant; no HCLPF capacities reported.

Table 5.4
PWR Seismic Success Paths

Plant Name: Turkey Point, Units 3 and 4   1 Sheet of 1

CHALLENGE	STRATEGY	SUCCESS PATH	RX		PRIMARY INTEGRITY					PRIMARY INVENTORY-INJECTION						PRIMARY INVENTORY-RECIRC					SECONDARY INTEGRITY					SECONDARY INVENTORY							CONTAINMENT									NOTES
			RPS	BI	PPORV	PSRV	PAD1	PAD2	RCPS	CHP1	HPI	LPI	ACC	AI1	AI2	CHPR	HPR	LPR	AR1	AR2	SGSS	SGA	TT	MSIV	TB	SG	MFW	NISP	AFW	AM1	AM2	AM3	CS1	CS2	FC1	FC2	IC	CI1	CI2	IGN	RF	
T-Loop		P		X																									X													

Challenge: One of the following: S1, S2, S3, A, V(-xx), T-LOOP, T-RX, T-TT, T-ATWS, T-UHS, T-RCP, T-LNMU, T-LMFW, T-EXFW, T-SLBOC, T-SLBIC, T-SGTR, T-SORV/IORV, T-SSI, T-(Other), OR T-(Support System). (-xx) refers to optional supplementary material.

Acronym of Support Systems: AC, ACBU1, ACBU2, ACBU3, AUXC2, AUXC3, AUXC4, CCW, DC, EAC, EDC, ESAS1, ESAS2, ESW, HVAC1, HVAC2, HVAC3, IA, NIT, OA3, 0A4, SA, STM, SW2, SW3, SW4, VAC

1,2,3...How many needed to operate   H = Human action required   T = Must be throttled/controlled

For Core Damage Prevention Challenges, show only hardware whose failure is modeled as contributing to core damage.

Table 5.5
PWR Accident Sequence Overview Table

Plant Name: Turkey Point, Units 3 and 4 For Fire PRA Only 1 Sheet of 1

#	Sequence	PDS	CDF	Init. Event	Lost Supports	Failed Functions	Attributes
1	Control Room		1.9×10^{-4}/ry	S3		RCS-INT, HPI, HPR	HUM
2	Cable Spreading Room		2.8×10^{-6}/ry	S3		RCS-INT, HPI, HPR	HUM
3	Zone 63 - Unit 3		2.7×10^{-6}/ry	S3		SSMU	HUM
4	Zone 61 - Unit 4		2.7×10^{-6}/ry	S3		SSMU	HUM

Init. Event (Initiator): One of the following: S1, S2, S3, A, V (-xx), T-LOOP, T-RX, T-TT, T-ATWS, T-UHS, T-RCP, T-LNMU, T-LMFW, T-EXFW, T-SLBOC, T-SLBIC, T-SGTR, T-SORV/IORV, T-SSI, T-(Other), or T-(Support System)
(-xx) refers to optional supplementary material.

Lost Supports: At most two of the following: AC, ACBU1, ACBU2, ACBU3, AUXC2, AUXC3, AUXC4, CCW, DC, EAC, EDC, ESAS1, ESAS2, ESW, HVAC1, HVAC2, HVAC3, IA, NIT, OA3, OA4, SA, STM, SW2, SW3, SW4, VAC (Field may be blank).

Failed Functions: At most three of the following: SINT, SDEP, SSMU, RCS-BOR, RCS-INT, RCS-DEP, HPI, HPR, LPI, LPR, CPSI, CPSR, CIF, VENT (If a 4th and/or 5th are necessary, use the "Notes" field)

Attributes: At most three of the following: ATWS, BYPASS, TIL, IND-SGTR, SBO, OR HUM (Field may be blank)

Table 5.6
PWR Accident Sequence Detailed Table

Plant Name: Turkey Point, Units 3 and 4 | For Fire PRA Only | 1 Sheet of 1

#	SEQUENCE	RX		PRIMARY INTEGRITY					PRIMARY INVENTORY-INJECTION						PRIMARY INVENTORY-RECIRC					SECONDARY INTEGRITY					SECONDARY INVENTORY							CONTAINMENT										NOTES
		RPS	BI	PPORV	PSRV	PAD1	PAD2	RCPS	CHP1	HPI	LPI	ACC	AI1	AI2	CHPR	HPR	LPR	AR1	AR2	SGS	SGA	TT	MSIV	TB	SG	MFW	NISP	AFW	AM1	AM2	AM3	CS1	CS2	FC1	FC2	IC	CI1	CI2	IGN	RF	HUM	
1	Control Room -1			X					X	X	X				X	X	X																								X	
2	Cable Spreading Room			X					X	X	X				X	X	X																								X	
3	Unit 3, Zone 63 - Control Rod Equipment Room								X	X																		X													X	
4	Unit 4, Zone 61 - Control Rod Equipment Room								X	X																		X													X	

6 REFERENCES

1. "Individual Plant Examination of External Events for Turkey Point Power Plant Units 3 and 4 in response to Generic Letter 88-20 Supplement 4," Florida Power and Light Company, June 1994.

2. "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10CFR50.54(f)," U. S. Nuclear Regulatory Commission Generic Letter 88-20, Supplement 4, June 28, 1991.

3. J. T. Chen, et. al, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," U. S. Nuclear Regulatory Commission, NUREG-1407, June 1991.

4. "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors," U. S. Nuclear Regulatory Commission Generic Letter 87-02, February 1987.

5. "Plant Specific Seismic Adequacy Evaluation of Turkey Point Units 3 and 4 to Resolve Unresolved Safety Issue (USI) A-46 and Generic Letter 87-02," prepared for Florida Power and Light Company by Stevenson and Associates, April 30, 1993.

6. "Evaluation of St. Lucie Unit 1, and Turkey Point Units 3 and 4, Resolution of Unresolved Safety Issue (USI) A-46, Supplement Number 1 to Generic Letter 87-02," U.S. Nuclear Regulatory Commission, Staff Evaluation Report, June 20, 1995.

7. "Supplemental Safety Evaluation Report of St. Lucie Unit 1 and Turkey Point Units 3 and 4, Resolution of Unresolved Safety Issue (USI) A-46," U.S. Nuclear Regulatory Commission internal memorandum from Richard H. Wessman, Chief of the Mechanical Engineering Branch, to Herbert N. Berkow, Director for Project Directorate II-2, September 19, 1996.

8. R. T. Sewell, et al., "Individual Plant Examination for External Events: Review Guidance," ERI/NRC 94-501 (Draft), May 1994.

9. "IPEEE Step 1 Review Guidance Document," U.S. Nuclear Regulatory Commission, June 18, 1992.

10. S. C. Lu and A. Boissonnade, "IPEEE Database Data Entry Sheet Package," Lawrence Livermore National Laboratory, December 14, 1993.

11. "Turkey Point Plant Units 3 and 4 Probabilistic Risk Assessment, Individual Plant Examination, Final Report," Florida Power and Light Company, June 1991.

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13. "Turkey Point Units 3 and 4, Docket Nos. 50-250 and 50-251, Response to Request for Additional Information, Generic Letter 88-20, Supplement 4 - 'Individual Plant Examination of External Events (IPEEE) for Sever Accident Vulnerabilities'," letter to U. S. Nuclear Regulatory Commission, from R. J. Hovey, Florida Power and Light Company, dated December 18, 1995.

14. Layout drawings from a fire hazard analysis report for Turkey Point Plant Units 3 and 4; Figure Numbers 9.6A-8 through 9.5A-15.

15. "Masonry Wall Design," U.S. Nuclear Regulatory Commission, IE Bulletin 80-11, 1980.

16. "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment," Seismic Qualification Utility Group (SQUG), Revision 2, February 14, 1992.

17. "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," Electric Power Research Institute, EPRI NP-6041-SL, Revision 1, August 1991.

18. "Use of Seismic Experience Data to Show Ruggedness of Equipment in Nuclear Power Plants (Draft)," Senior Seismic Review Advisory Panel, January 17, 1990.

19. "Fire Induced Vulnerability Evaluation (FIVE)," Electric Power Research Institute, TR-100370, Revision 0, November 2, 1990.

20. "Shutdown Decay Heat Removal Analysis of a Westinghouse Pressurized Water Reactor," U.S. Nuclear Regulatory Commission, NUREG/CR-4762, March 1987.

21. "Oconee Probabilistic Risk Assessment," Duke Power Company and Nuclear Safety Analysis Center/EPRI, NSAC/60, June 1984.

22. "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," Regulatory Guide 1.95, February 1975.

23. "Staff Guidance of IPEEE Submittal Review on Resolution of Generic or Unresolved Safety Issues (GSI/USI)," U.S. Nuclear Regulatory Commission, August 21, 1997.

RISK-INFORMED INSPECTION NOTEBOOK FOR

TURKEY POINT NUCLEAR PLANT

UNITS 3 AND 4

PWR, WESTINGHOUSE, THREE-LOOP PLANT WITH LARGE DRY CONTAINMENT

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Enclosure

NOTICE

This notebook was developed for the NRC's inspection teams to support risk-informed inspections. The activities involved in these inspections are discussed in "Reactor Oversight Process Improvement," SECY-99-007A, March 1999. The user of this notebook is assumed to be an inspector with an extensive understanding of plant-specific design features and operation. Therefore, the notebook is not a stand-alone document, and may not be suitable for use by non-specialists. This notebook will be periodically updated with new or replacement pages incorporating additional information on this plant. Technical errors in, and recommended updates to, this document should be brought to the attention of the following person:

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ABSTRACT

This notebook contains summary information to support the Significance Determination Process (SDP) in risk-informed inspections for Turkey Point Nuclear Plant Units 3 and 4.

SDP worksheets support the significance determination process in risk-informed inspections and are intended to be used by the NRC's inspectors in identifying the significance of their findings, i.e., in screening risk-significant findings, consistent with Phase-2 screening in SECY-99-007A. To support the SDP, additional information is given in an Initiators and System Dependency table, and as simplified event-trees, called SDP event-trees, developed in preparing the SDP worksheets.

The information contained herein is based on the licensee's IPE submittal. The information is revised based on IPE updates or other licensee or review comments providing updated information and/or additional details.

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TABLES

1. INFORMATION SUPPORTING SIGNIFICANCE DETERMINATION PROCESS (SDP)

SECY-99-007A (NRC, March 1999) describes the process for making a Phase-2 evaluation of the inspection findings. In Phase 2, the first step is to identify the pertinent core damage scenarios that require further evaluation based on the specifics of the inspection findings. To aid in this process, this notebook provides the following information:

1. Initiator and System Dependency Table
2. Significance Determination Process (SDP) Worksheets
3. SDP Event Trees

The initiator and system dependency table shows the major dependencies between front-line- and support-systems, and identifies their involvement in different types of initiators. The information in this table identifies the most risk-significant front-line- and support-systems; it is not an exhaustive nor comprehensive compilation of the dependency matrix as known in Probabilistic Risk Assessments (PRAs). For pressurized water reactors (PWRs), the support systems for Reactor Coolant Pump (RCP) seals are explicitly denoted to assure that the inspection findings on them are properly accounted for. This table is used to identify the SDP worksheets to be evaluated, corresponding to the inspection's findings on systems and components.

To evaluate the impact of the inspection's finding on the core-damage scenarios, the SDP worksheets are developed and provided. They contain two parts. The first part identifies the functions, the systems, or combinations thereof that can perform mitigating functions, the number of trains in each system, and the number of trains required (success criteria) for each class of initiators. The second part of the SDP worksheet contains the core-damage accident sequences associated with each initiator class; these sequences are based on SDP event trees. In the parenthesis next to each of the sequence the corresponding event tree branch number(s) representing the sequence is included. Multiple branch numbers indicate that the different accident sequences identified by the event tree are merged into one through the boolean reduction. The classes of initiators that are considered in this notebook are 1) Transients, 2) Small Loss of Coolant Accident (LOCA), 3) Stuck-open Power Operated Relief Valve (PORV), 4) Medium LOCA, 5) Large LOCA, 6) Loss of Offsite Power (LOOP), 7) Steam Generator Tube Rupture (SGTR), and 8) Anticipated Transients Without Scram (ATWS). Main Steam Line Break (MSLB) events are included separately if they are treated as such in the licensee's Individual Plant Examination (IPE) submittal.

Following the SDP worksheets, the SDP event trees corresponding to each of the worksheets are presented. The SDP event trees are simplified event trees developed to define the accident sequences identified in the SDP worksheets.

The following items were considered in establishing the SDP event trees and the core-damage sequences in the SDP worksheets:

1. Event trees and sequences were developed such that the worksheet contains all the major accident sequences identified by the plant-specific IPEs. In cases where a plant-specific feature introduced a sequence that is not fully captured by our existing set of initiators and event trees, then a separate worksheet is included.

2. The event trees and sequences for each plant took into account the IPE models and event trees for all similar plants. Any major deviations in one plant from similar plants typically are noted at the end of the worksheet.

3. The event trees and the sequences were designed to capture core-damage scenarios, without including containment-failure probabilities and consequences. Therefore, branches of event trees that are only for the purpose of a Level II PRA analysis are not considered. The resulting sequences are merged using Boolean logic.

4. The simplified event-trees focus on classes of initiators, as defined above. In so doing, many separate event trees in the IPEs often are represented by a single tree. For example, some IPEs define four classes of LOCAs rather than the three classes considered here. The sizes of LOCAs for which high-pressure injection is not required are some times divided into two classes, the only difference between them being the need for reactor scram in the smaller break size. Some IPEs also may define several classes of transients, depending on the initiator's impact on the systems. Such differentiations generally are not considered in the SDP worksheets unless they could not be accounted for by the Initiator and System Dependency table.

5. Major operator actions during accident scenarios are assigned as high stress operator action or an operator action using simple, standard criteria among a class of plants. This approach resulted in the designation of some operator actions as high-stress ones (as opposed to normal), even though the PRA may have assumed a (routine) operator action; hence, they have been assigned an error probability less than 5E-2 in the IPE. In such cases, a note is given at the end of the worksheet.

The three sections that follow include the initiators and dependency table, SDP worksheets, and the SDP event-trees for the Turkey Point Nuclear Plant Units 3 and 4.

1.1 INITIATORS AND SYSTEM DEPENDENCY

Table 1 provides the list of the systems included in the SDP worksheets, the major components in the systems, and the support system dependencies. The system involvements in different initiating events are noted in the last column.

Table 1 Initiators and System Dependency for Turkey Point Units 3 & 4

Affected System		Major Components	Support Systems	Initiating Event Scenarios
Code	Name			
ACC	Accumulators	Three accumulators		LLOCA
AFW	Auxiliary Feedwater System	Three TDPs	EPS (3, 4)[1], DC, ESF (3,4), IA	Transient, SLOCA, LOOP, SGTR, ATWS
CCW	Component Cooling Water System	Two cooling loops	EPS (3, 4), DC, ESF (3,4), ICW (3, 4)	Transient, SLOCA, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP Seal LOCA
CIS	Containment Isolation System	Containment isolation valves	EPS (3, 4), DC, ESF (3,4)	
CSS	Containment Spray System	Two pump trains	LHSI/RHR (3, 4), EPS (3, 4), DC, ESF (3,4), CCW (3, 4), HVAC	Transient, SLOCA, MLOCA, LLOCA, LOOP, SGTR
CVCS	Chemical and Volume Control System	Three charging pumps and two boric acid transfer pumps	EPS (3, 4), DC, CCW (3, 4), IA, HVAC	Transient, LOOP, ATWS, RCP Seal LOCA
CVHRS	Containment Ventilation and Heat Removal System	Three Emergency Containment Coolers	EPS (3, 4), DC, ESF (3,4), CCW (3, 4)	Transient, SLOCA, MLOCA, LLOCA, LOOP, SGTR
EPS	Electric Power System (Power Generation and AC and DC Power Distribution)	Four 4.16 kV buses with two EDGs	DC, ESF (3,4), IA (only EPS3), HVAC	Transient, SLOCA, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP Seal LOCA
		Four 125 VDC buses shared by both units	AC (3, 4)	Transient, SLOCA, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP Seal LOCA

Table 1 (Continued)

Affected System		Major Components	Support Systems	Initiating Event Scenarios
Code	Name			
ESF/RPS	Engineered Safeguard Feature Actuation System / Reactor Protection System	Protection and safeguards logic cabinets	ESF: DC, HVAC RPS: DC	Transient, SLOCA, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP Seal LOCA
HHSI	High Head Safety Injection	Four HHSI pumps (two per unit)	CSS (3, 4), CVHRS (3, 4), LHSI/RHR (3, 4), EPS (3, 4), DC, ESF (3,4), CCW (3, 4), HVAC	Transient, SLOCA, MLOCA, LOOP, SGTR
HVAC	Heating, Ventilation and Air Conditioning System	Several independent subsystems	EPS (3, 4), ESF (3,4)	Transient, SLOCA, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP Seal LOCA
IA	Instrument Air System	Two diesel compressors		Transient, SLOCA, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP Seal LOCA
ICW	Intake Cooling Water System	Three ICW pumps	EPS (3, 4), DC, ESF (3,4)	Transient, SLOCA, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP Seal LOCA
LHSI / RHR	Low Head Safety Injection / Residual Heat Removal System	Two LHSI/RHR pumps per unit (1 Multi-Train System)	CSS (3, 4), CVHRS (3, 4), EPS (3, 4), DC, ESF (3,4), CCW (3, 4), IA, HVAC	Transient, SLOCA, MLOCA, LLOCA, LOOP, SGTR
PCS	Power Conversion System	Two MFW pumps, three ADVs, four SCDVs, three condensate pumps, two standby FW pumps	EPS (3, 4), DC, ESF (3,4), ICW (3, 4), IA	Transient
PPC	Primary Pressure Control System	Two PORVs, three Code Safety Valves	EPS (3, 4), DC, IA	Transient, SLOCA, LOOP, SGTR, ATWS

Table 1 (Continued)

Affected System		Major Components	Support Systems	Initiating Event Scenarios
Code	Name			
RCP	Reactor Coolant Pumps	Seals	1 / 3 CVCS trains in seal injection (1 multi-train system) or 1 / 2 CCW trains to thermal barrier in RCPs (1 multi-train system)	Transient, LOOP, RCP Seal LOCA
V	Interfacing Systems LOCA / Containment Bypass	Four penetrations: 1, 2, 11, 43		Interfacing Systems LOCA

Notes:

(1) (3, 4) means that a system in Unit 3 will be supported by a support system in Unit 3; the same applies to Unit 4. For example, the CVCS of Unit 3 is supported by the EPS of Unit 3, and the CVCS of Unit 4 is supported by the EPS of Unit 4.

(2) CDF of a single unit: 1.0E-4/reactor year. In the SDP Worksheets, the success criteria are per unit, except where a dual-unit initiator is noted.

1.2 SDP WORKSHEETS

This section presents the SDP worksheets to be used in the Phase 2 evaluation of the inspection findings for the Turkey Point Nuclear Plant Units 3 & 4. The SDP worksheets are presented for the following initiating event categories:

1. Transients
2. Small LOCA
3. Stuck-open PORV
4. Medium LOCA
5. Large LOCA
6. LOOP
7. Steam Generator Tube Rupture (SGTR)
8. Anticipated Transients Without Scram (ATWS)
9. Special Initiators

Table 2.1 SDP Worksheet for Turkey Point Nuclear Plant Units 3 & 4 Transients

Estimated Frequency (Table 1 Row) ________ Exposure Time ________ Table 1 Result (circle): A B C D E F G H

Safety Functions Needed:	***Full* Creditable Mitigation Capability for Each Safety Function:**
Power Conversion System (PCS)	Operator restores feedwater to SGs using 1 / 2 Main Feedwater pumps or 1 / 2 Standby SG Feedwater pumps (operator action)
Secondary Heat Removal (AFW)	1/3 TDP trains of AFW (375 gpm) (1 multi-train system)
Early Inventory, High Pressure Injection (EIHP)	2 / 4 HHSI pump trains (1 multi-train system)
Primary Heat Removal, Feed/Bleed (FB)	2 / 2 PORVs (Operator action)
High Pressure Recirculation (HPR)	1 / 4 HHSI pump trains with 1/ 2 RHR pump trains (Operator action for switchover = operator action)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 TRANS - PCS - AFW - HPR (4)			
2 TRANS - PCS - AFW - FB (5)			
3 TRANS - PCS - AFW - EIHP (6)			

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:
If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Table 2.2 SDP Worksheet for Turkey Point Nuclear Plant Units 3 & 4 Small LOCA

Estimated Frequency (Table 1 Row) ________ Exposure Time ________ Table 1 Result (circle): A B C D E F G H

Safety Functions Needed:	***Full* Creditable Mitigation Capability for Each Safety Function:**
Secondary Heat Removal (AFW)	1/3 TDP trains of AFW (375 gpm) (1 multi-train system)
Early Inventory, HP Injection (EIHP)	2 / 4 HHSI pump trains (1 multi-train system)
RCS Cooldown/ Depressurization (RCSDEP)	Operator depressurizes RCS using 1 / 2 PORVs (operator action)
Primary Heat Removal (F&B)	2 / 2 PORVs (operator action)
Low Pressure Injection (LPI)	1 / 2 RHR pump trains (1 multi-train system)
High Pressure Recirculation (HPR)	1 / 4 HHSI pump trains with 1/ 2 RHR pump trains (Early Operator action to shut down the RHR pumps and switchover from injection to recirculation = operator action)
Low Pressure Recirculation (LPR)	1/ 2 RHR pump trains (Operator action to switchover = operator action)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 SLOCA - HPR - LPR (3)			
2 SLOCA - AFW - HPR (5)			

3 SLOCA - AFW - F&B (6)			
4 SLOCA - EIHP - LPR (8)			
5 SLOCA - EIHP - LPI (9)			
6 SLOCA - EIHP - RCSDEP (10)			
7 SLOCA - EIHP - AFW (11)			

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Table 2.3 SDP Worksheet for Turkey Point Nuclear Plant Units 3 & 4 — Stuck Open PORV (SORV)

Estimated Frequency (Table 1 Row) ________ Exposure Time ________ Table 1 Result (circle): A B C D E F G H

Safety Functions Needed:	*Full* Creditable Mitigation Capability for Each Safety Function:
Secondary Heat Removal (AFW)	1/3 TDP trains of AFW (375 gpm) (1 multi-train system)
Isolation of Small LOCA (BLK)	The closure of the block valve associated with stuck open PORV (recovery action)
Early Inventory, HP Injection (EIHP)	2 / 4 HHSI pump trains (1 multi-train system)
RCS Cooldown / Depressurization (RCSDEP)	Operator depressurizes RCS using 1 / 2 PORVs (Operator action)
Primary Heat Removal (F&B)	2 / 2 PORVs (Operatior action)
Low Pressure Injection (LPI)	1 / 2 RHR pump trains (1 multi-train system)
High Pressure Recirculation (HPR)	1 / 4 HHSI pump trains with 1/ 2 RHR pump trains (Early Operator action to shut down the RHR pumps and switchover from injection to recirculation = operator action)
Low Pressure Recirculation (LPR)	1 / 2 RHR pump trains (Operator action for switchover = operator action)

Circle Affected Functions	Recovery of Failed Train	*Remaining Mitigation Capability Rating for Each Affected Sequence*	Sequence Color
1 SORV - BLK - HPR - LPR (3)			
2 SORV - BLK - AFW - HPR (5)			
3 SORV - BLK - AFW - F&B (6)			
4 SORV - BLK - EIHP - LPR (8)			
5 SORV - BLK - EIHP - LPI (9)			

6 SORV - BLK - EIHP - RCSDEP (10)			
7 SORV - BLK - EIHP - AFW (11)			

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Table 2.4 SDP Worksheet for Turkey Point Nuclear Plant Units 3 & 4 — Medium LOCA

Estimated Frequency (Table 1 Row) ________	Exposure Time ________	Table 1 Result (circle): A B C D E F G H	
Safety Functions Needed:	***Full* Creditable Mitigation Capability for Each Safety Function:**		
Early Inventory, HP Injection (EIHP)	2 / 4 HHSI pump trains (1 multi-train system)		
Low Pressure Recirculation (LPR)	1 / 2 RHR pump trains (Operator switchover from injection to recirculation = operator action)		
Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 MLOCA - LPR (2)			
2 MLOCA - EIHP (3)			

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Table 2.5 SDP Worksheet for Turkey Point Nuclear Plant Units 3 & 4 — Large LOCA

Estimated Frequency (Table 1 Row) ________ Exposure Time ________ Table 1 Result (circle): A B C D E F G H

Safety Functions Needed:	*Full* Creditable Mitigation Capability for Each Safety Function:
Early Inventory, Accumulators (EIAC)	2 / 2 accumulators (1 train)(1)
Early Inventory, LP Injection (EILP)	1 / 2 pumps LHSI/RHR pump trains (1 multi-train system)
Low Pressure Recirculation (LPR)	1/2 pumps LHSI/RHR pump trains with operator switchover from injection to recirculation (operator action under high stress)(2)
Containment Pressure / Temperature Control (CNT)	1 / 2 pump trains of CSS with 2 / 3 Emergency Containment Coolers (1 multi-train system)

Circle Affected Functions	Recovery of Failed Train	*Remaining Mitigation Capability Rating for Each Affected Sequence*	Sequence Color
1 LLOCA - CNT (2)			
2 LLOCA - LPR (3)			
3 LLOCA - EILP (4)			
4 LLOCA - EIAC (5)			

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event: If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Notes:

(1) Accumulators are passive, highly reliable components and their probability of failure may be smaller than 1E-2.

(2) The human error probability assessed by the IPE (page 3.0-217) is 1.2E-1 (event U30PALPR).

Table 2.6 SDP Worksheet for Turkey Point Nuclear Plant Units 3 & 4 LOOP

Estimated Frequency (Table 1 Row) ________ Exposure Time ________ Table 1 Result (circle): A B C D E F G H

Safety Functions Needed:	***Full* Creditable Mitigation Capability for Each Safety Function:**
Emergency AC Power (EAC)	1 / 2 Emergency Diesel Generators (1 multi-train system)
Turbine-driven AFW pump (TDAFW)	1 / 3 TDP trains of AFW (375 gpm) (1 multi-train system)
Recovery of AC Power in < 2 hrs (REC2)	SBO procedures implemented (Operator action under high stress)(1)
Recovery of AC Power in < 5 hrs (REC5)	SBO procedures implemented (Operator action)(2, 3)
Early Inventory, HP Injection (EIHP)	2 / 4 HHSI pump trains (1 multi-train system)
Primary Heat Removal (FB)	2 / 2 PORVs (operator action)
High Pressure Recirculation (HPR)	1 / 4 HHSI pump trains with 1/ 2 RHR pump trains and with operator action for switchover (operator action)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 LOOP - TDAFW - HPR (3, 11)			
2 LOOP - TDAFW - FB (4, 12)			
3 LOOP - TDAFW - EIHP (5, 13)			
4 LOOP - EAC - HPR (7) (AC recovered)			

5 LOOP - EAC - EIHP (8) (AC recovered)			
6 LOOP - EAC - REC5 (9)			
7 LOOP - EAC - TDAFW - REC2 (14)			

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Notes:

(1) The IPE's human action that is similar to "Recovery of AC Power in < 2 hrs (REC2)": is "LOOP with AFW failure" (RU3DT1D4-1), and it has a human error probability equal to 8.5E-2.

(2) The IPE's human action that is similar to "Recovery of AC Power in < 5 hrs (REC5)" is "Offsite Power Restoration Prior to Battery Depletion" (RU3BATDEP), and it has a human error probability equal to 1.0E-2.

(3) In an SBO situation, an RCP seal LOCA may occur, with subsequent core damage at about 5 hours.

Table 2.7 SDP Worksheet for Turkey Point Nuclear Plant Units 3 & 4 SGTR

Estimated Frequency (Table 1 Row) ________ Exposure Time ________ Table 1 Result (circle): A B C D E F G H

Safety Functions Needed:	*Full* Creditable Mitigation Capability for Each Safety Function:
Secondary Heat Removal (SHR)	1 / 3 TDP trains of AFW (375 gpm) (1 multi-train system)
Early Inventory, HP Injection (EIHP)	2 / 4 HHSI pump trains (1 multi-train system)
Pressure Equalization (EQ)	Operator isolates ruptured SG and depressurizes RCS to less than setpoint of relief valves of SG (operator action under high stress[1]
Feed-and-Bleed (FB)	2 / 2 PORVs (operator action)
High Pressure Recirculation (HPR)	1 / 4 HHSI pump trains (1 multi-train system) with 1/ 2 RHR pump trains and with operator switchover to recirculation (operator action)
Long-Term RCS Makeup Source (LTMS)	Operator refills RWST (operator action)

Circle Affected Functions	Recovery of Failed Train	*Remaining Mitigation Capability Rating for Each Affected Sequence*	Sequence Color
1 SGTR - EQ - LTMS (3)			
2 SGTR - EIHP - EQ (5)			
3 SGTR - SHR - HPR - LTMS (8)			
4 SGTR - SHR - FB (9)			
5 SGTR - SHR - EQ (10)			

6 SGTR - SHR - EIHP (11)			
Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event: If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.			

<u>Note</u>:

(1) Operator isolates the ruptured SG and depressurizes RCS is represented in the IPE (page 3-217) by two human actions: 1) "Operator Fails to Control SG Level (Overfill)" (event AHFF3SGLC, human error probability (HEP) = 7.5E-5), and 2) "Failue to Depressurize to Reduce Primary / Secondary Leak (SGTR)" (Event U3OPRDPZ, negligible HEP).

Table 2.8 SDP Worksheet for Turkey Point Nuclear Plant Units 3 & 4 ATWS

Estimated Frequency (Table 1 Row) _________ Exposure Time _________ Table 1 Result (circle): A B C D E F G H

Safety Functions Needed:	**Full Creditable Mitigation Capability for Each Safety Function:**
Turbine trip (TTP)	Operator trips the turbine or closes MSIVs (operator action)
Emergency Boration (EB)	Operator conducts emergency boration using 3 / 3 CVCS pump trains with 1 / 2 boric acid (operator action)
Secondary Heat Removal (AFW)	2 / 3 TDP trains of AFW (750 gpm) (1 train system)
Primary Relief (SRV)	3 / 3 SRVs or (2 / 3 SRVs and 2 / 2 PORVs) open (1 train)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 ATWS - SRV (3)			
2 ATWS - AFW (4)			
3 ATWS - EB (5)			
4 ATWS - TTP (6)			

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Table 2.9 SDP Worksheet for Turkey Point Nuclear Plant Units 3 & 4 — Special Initiators

Estimated Frequency (Table 1 Row) ________ Exposure Time ________ Table 1 Result (circle): A B C D E F G H

<u>Safety Functions Needed</u>:	<u>*Full* Creditable Mitigation Capability for Each Safety Function</u>:
"B" charging pump (CHB)	"B" charging pump providing RCP seal injection (operator action under high stress)[1]
Component Cooling Water (CCW)	1 / 2 cooling loops (1 train)[2]
Valve FCV-626 (FCVFAC)	Valve FCV-626 automatically closes on a high flow signal from flow instrument FIC-626 (1 train)
Isolate ISLOCA (OPFTCMGV)	Operator isolates ISLOCA by locally closing a manual gate valve (operator action)

<u>Circle Affected Functions</u>	<u>Recovery of Failed Train</u>	<u>*Remaining Mitigation Capability Rating for Each Affected Sequence*</u>	<u>Sequence Color</u>
Initiator: Loss of CCW (LOSSCCW)[3] (transient-induced LOCA)[4] 1 LOSSCCW - CHB (Dom 1)			
Initiator: Loss of Grid (LOSSGRID) (dual-unit initiator: transient-induced LOCA)[5] 2 LOSSGRID - CCW - CHB (Dom 2)			
Initiator: Interfacing system LOCA (ISLOCA: RCP thermal barrier heat exchanger tube rupture) 3 ISLOCA - FCVFAC - OPFTCMGV (Dom			
Initiator: Interfacing system LOCA in Penetration 11: failure of two in-series check valves (*-875A, B or C and *876D or E) (TWOCKVLV) 4 TWOCKVLV[5] (Dom 16)			

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Notes:

(1) The IPE assesses the probability of "Charging pump B out due to maintenance" equal to 4.79E-2 (Table 3.3-5, page 3.0-192)

(2) 1 train selected for event CCW to approximate the frequency of sequence 2 (4.69E-5/reactor year).

(3) In sequence 1, in addition of the initiator loss of CCW (LOSSCCW), other initiators leading to transient-induced LOCAs are: loss of DC bus, loss of 4.16 kV bus, loss of Instrument Air, loss of Intake Cooling Water, Feedline break and loss of Vital Instrument Panels (those panels whose loss will not initiate SI).

(4) A total loss of Component Cooling Water causes an RCP seal LOCA which, in turn, causes core damage. "B" charging pump can provide RCP seal injection independent of the CCW/ICW system.

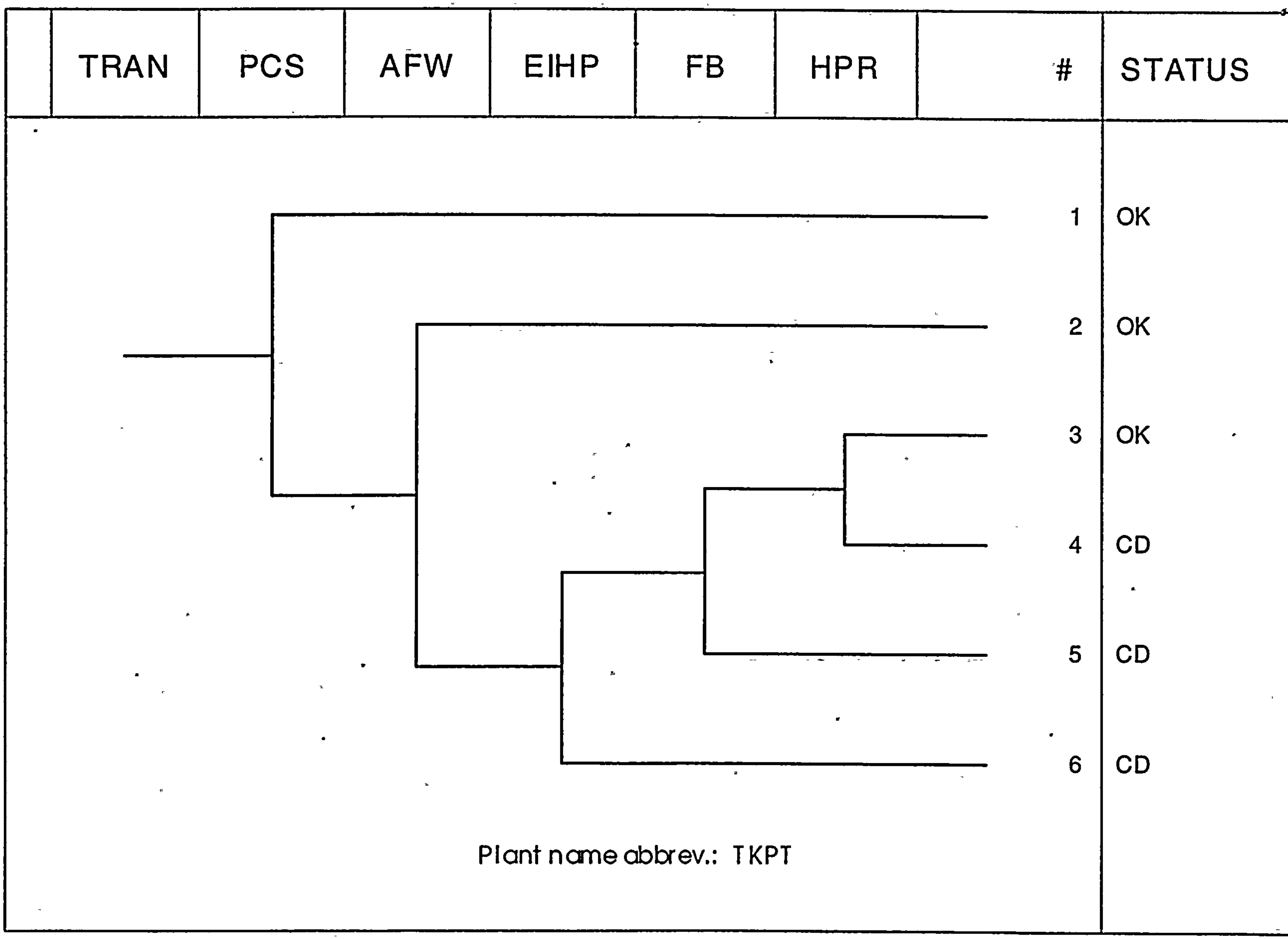
(5) The IPE assesses a frequency of 2.0E-6/reactor year for an interfacing system LOCA in penetration 11.

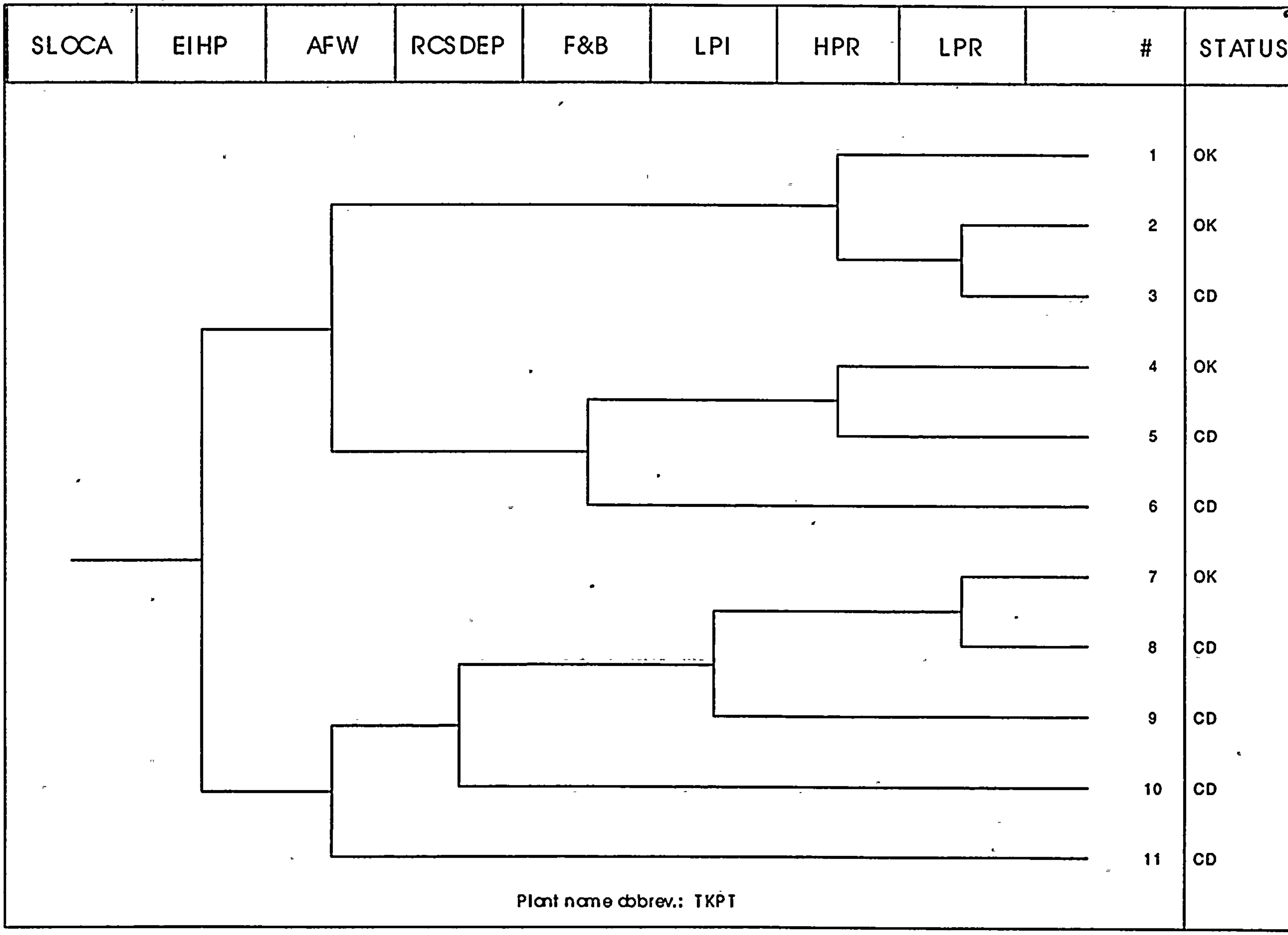
1.3 SDP Event Trees

This section provides the simplified event trees called SDP event trees used to define the accident sequences identified in the SDP worksheets in the previous section. An event tree for the stuck-open PORV is not included since it is similar to the small LOCA event tree. The event tree headings are defined in the corresponding SDP worksheets.

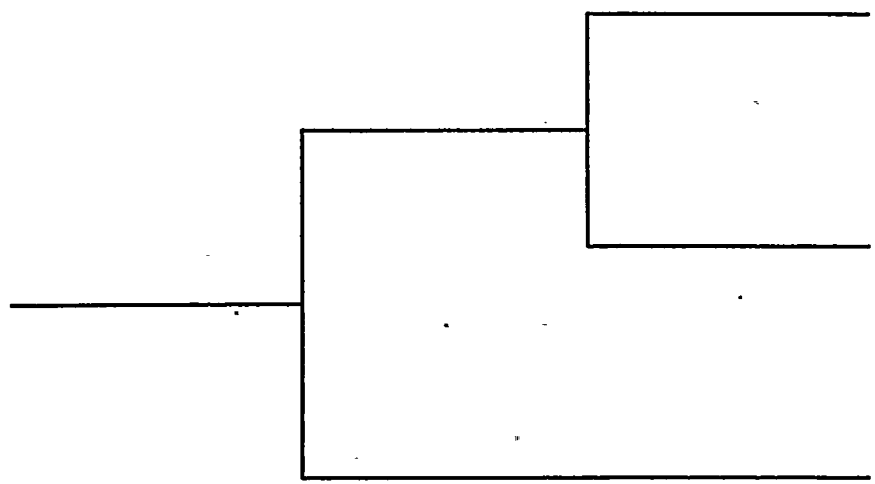
The following event trees are included:

1. Transients
2. Small LOCA
3. Medium LOCA
4. Large LOCA
5. LOOP
6. Steam Generator Tube Rupture (SGTR)
7. Anticipated Transients Without Scram (ATWS)

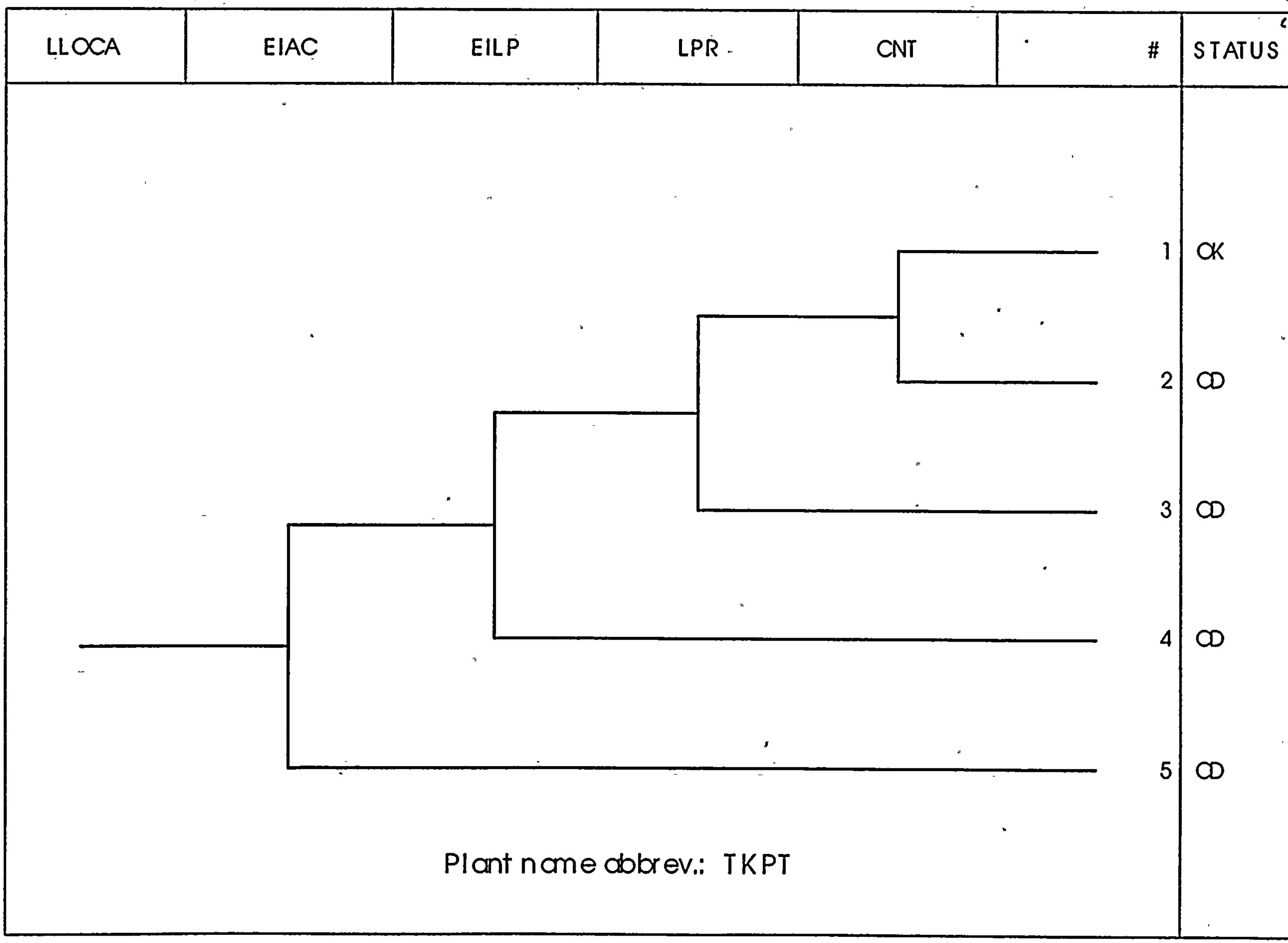

TRAN
PCS
AFW
EIHP
FB
HPR
#
STATUS
1
OK
2
OK
3
OK
4
CD
5
CD
6
CD
Plant name abbrev.: TKPT

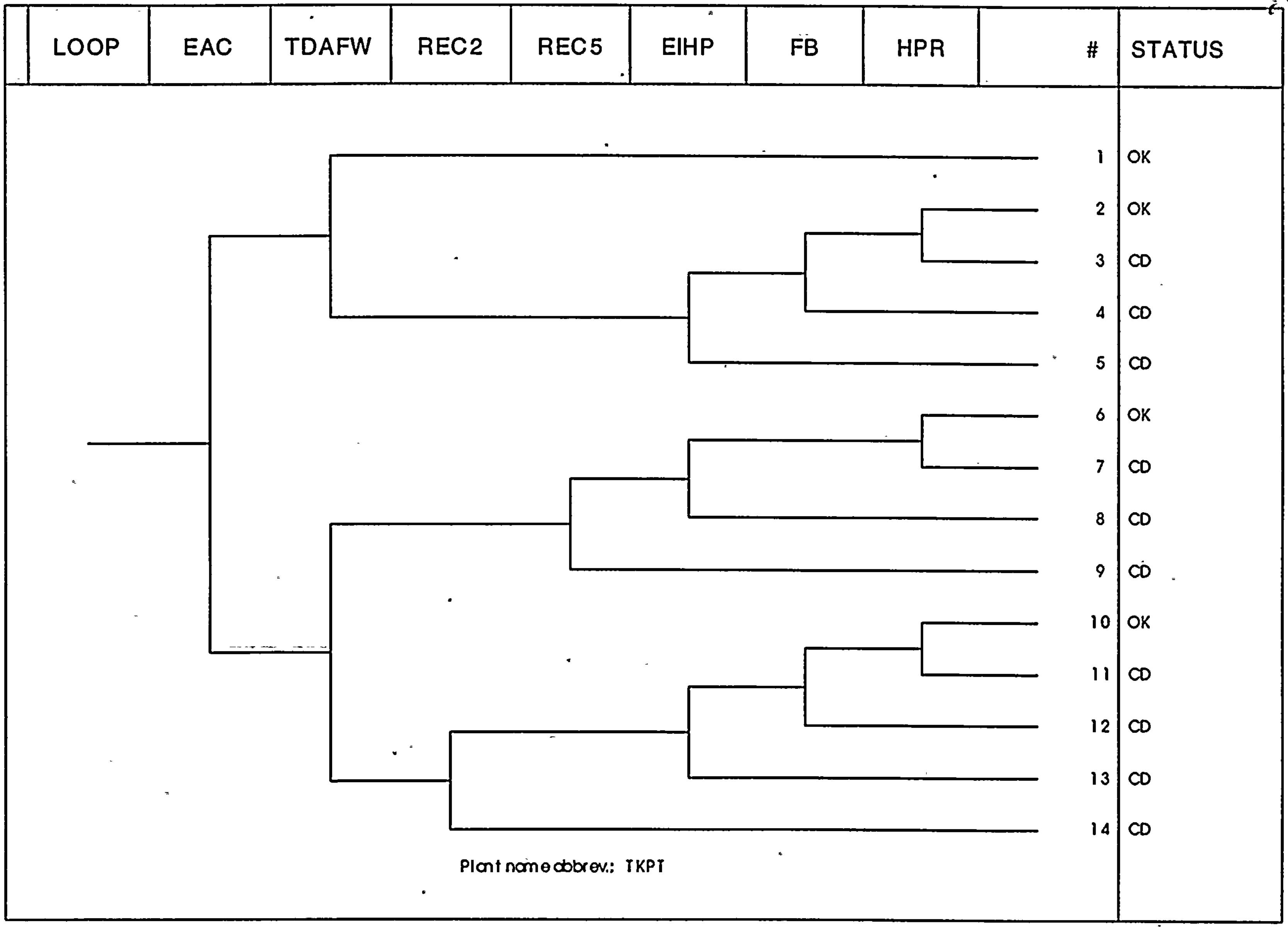

SLOCA
EIHP
AFW
RCSDEP
F&B
LPI
HPR
LPR
#
STATUS
1 OK
2 OK
3 CD
4 OK
5 CD
6 CD
7 OK
8 CD
9 CD
10 CD
11 CD
Plant name abbrev.: TKPT

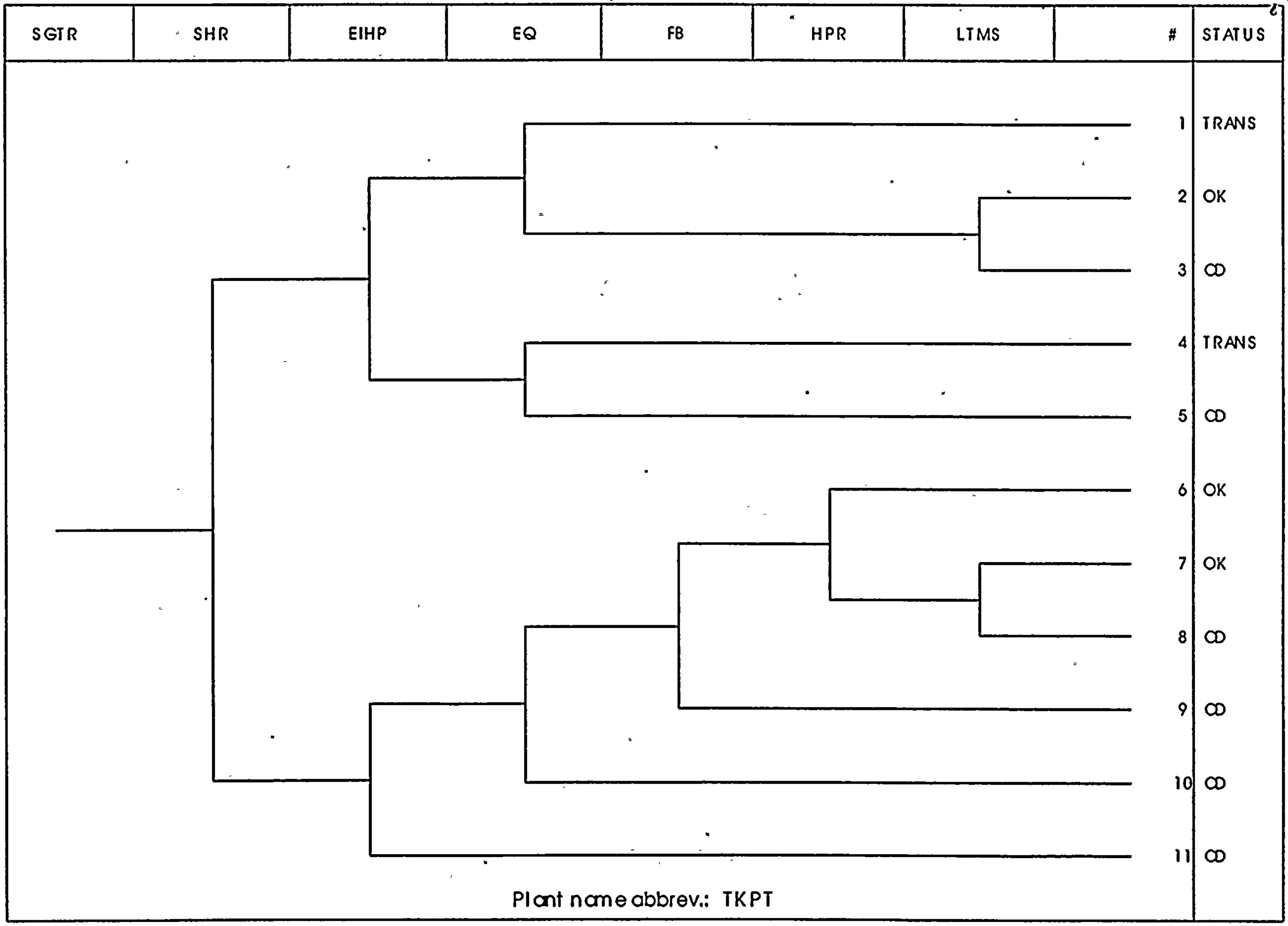
	MLOCA	EIHP	LPR	#	STATUS
				1	OK
				2	CD
				3	CD

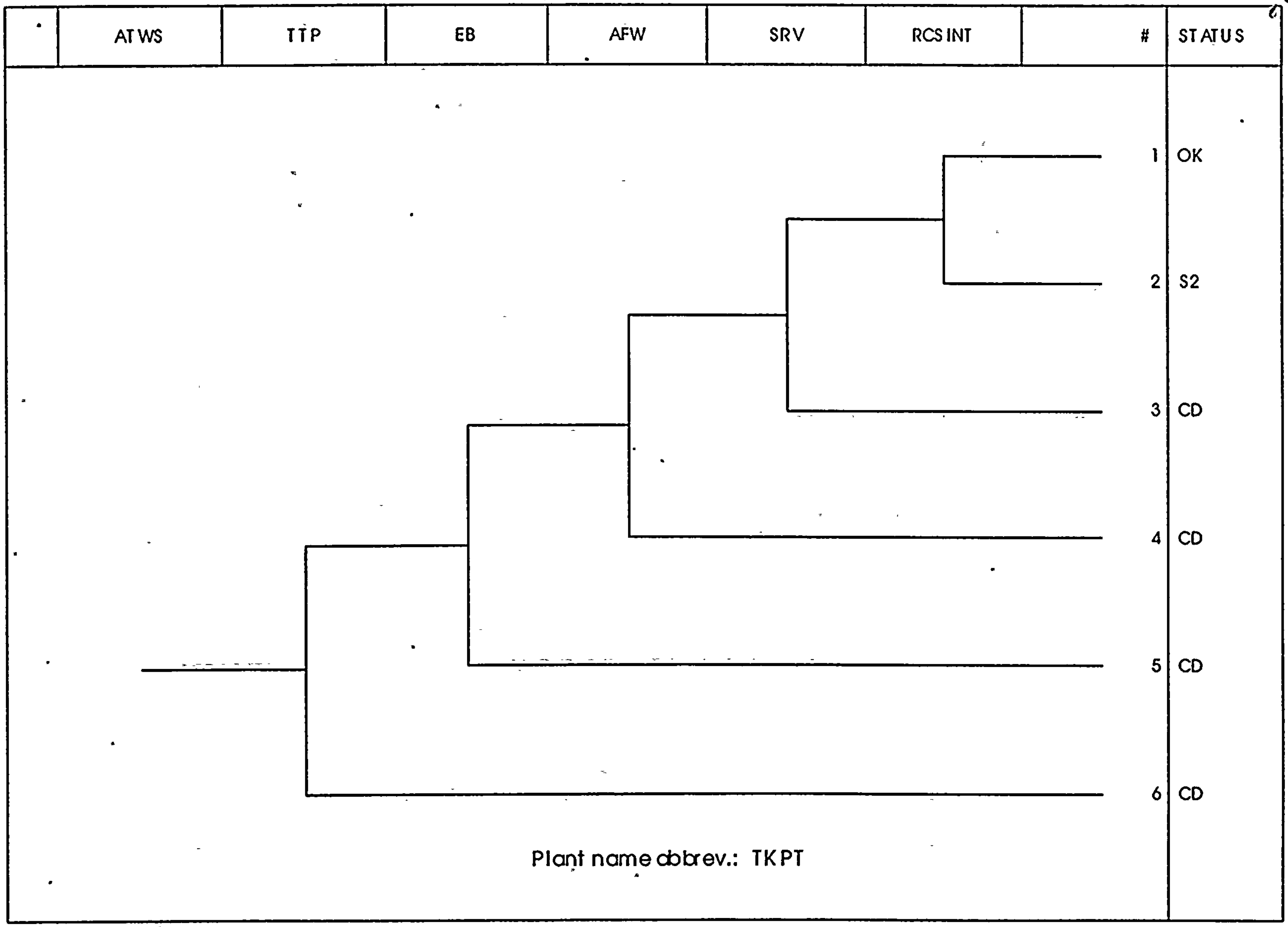


Plant name abbrev.: TKPT


LLOCA
EIAC
EILP
LPR
CNT
#
STATUS
1
OK
2
CD
3
CD
4
CD
5
CD
Plant name abbrev.: TKPT


LOOP
EAC
TDAFW
REC2
REC5
EIHP
FB
HPR
#
STATUS
1 OK
2 OK
3 CD
4 CD
5 CD
6 OK
7 CD
8 CD
9 CD
10 OK
11 CD
12 CD
13 CD
14 CD
Plant name abbrev.: TKPT


SGTR
SHR
EIHP
EQ
FB
HPR
LTMS
#
STATUS
1 TRANS
2 OK
3 CD
4 TRANS
5 CD
6 OK
7 OK
8 CD
9 CD
10 CD
11 CD
Plant name abbrev.: TKPT


ATWS
TTP
EB
AFW
SRV
RCS INT
#
STATUS
1 OK
2 S2
3 CD
4 CD
5 CD
6 CD
Plant name abbrev.: TKPT

2. RESOLUTION AND DISPOSITION OF COMMENTS

This section documents the comments received on the material included in this report and their resolution. This section is blank until comments are received and are addressed.

REFERENCES

1. NRC SECY-99-007A, Recommendations for Reactor Oversight Process Improvements (Follow-up to SECY-99-007), March 22, 1999.

2. Florida Power & Light Company, "Turkey Point, Units 3 & 4 – Individual Plant Examination Report," June 25, 1991.

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