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

May 9, 1994

Docket No. 50-333

Mr. William A. Josiger, Acting
Executive Vice President, Nuclear
Generation
Power Authority of the State of
New York
123 Main Street
White Plains, New York 10601

Dear Mr. Josiger:

SUBJECT: INDIVIDUAL PLANT EXAMINATION FOR THE JAMES A. FITZPATRICK NUCLEAR POWER PLANT (TAC NO. M74411)

On November 23, 1988, the NRC issued Generic Letter (GL) 88-20 which requires licensees to conduct an Individual Plant Examination (IPE) in order to identify potential severe accident vulnerabilities at their plant, and report the results to the Commission. Through the examination process, a licensee is expected to: (1) develop an overall appreciation of severe accident behavior; (2) understand the most likely severe accident sequences that could occur at its plant; (3) gain a more quantitative understanding of the overall probabilities of core damage and fission product releases; and, (4) if necessary, reduce the overall probability of core damage and radioactive material releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

As stated in Appendix D of the IPE submittal guidance document NUREG-1335, all IPEs are to be reviewed by NRC teams to determine the extent to which each licensee's IPE process met the intent of GL 88-20. The IPE review itself is a two step process; the first step, or "step 1" review, focuses on completeness and the quality of the submittal. Only selected IPE submittals, determined on a case-by-case basis, will be investigated in more detail under a second step or "step 2" review. The decision to go to a "step 2" review is primarily based on the ability of the licensee's methodology to identify vulnerabilities, and the consistency of the licensee's IPE findings and conclusions with previous probabilistic safety assessment (PSA) experience. A unique design may also warrant a "step 2" review to better understand the implication of certain IPE findings and conclusions.

On September 13, 1991, as supplemented by letters dated May 28, 1992, and September 1, 1992, the Power Authority of the State of New York (PASNY) submitted the FitzPatrick IPE in response to GL 88-20 and associated supplements. The IPE submittal is based on an internal events level 1 Probabilistic Risk Assessment (PRA), and a level 2 containment performance assessment consistent with the guidance provided in GL 88-20, Appendix 1. The IPE process also addressed internal flooding. PASNY plans to provide a

DF01

separate submittal for external events (IPEEE), which will be reviewed separately within the framework prescribed in GL 88-20, Supplement 4.

The FitzPatrick IPE is based on a level 1 PRA and a containment performance analysis consistent with GL 88-20, Appendix 1, guidelines. NUREG-1150 (Peach Bottom) insights and methodology were utilized extensively, and differences between the plants were accounted for in the models. PASNY personnel maintained involvement throughout the development and application of PRA techniques to the FitzPatrick facility. Outside consultants (Science Application International Corporation (SAIC) and Risk Management Associates) provided technical support, primarily by providing expertise in specific areas, e.g., human failure data analysis, common cause data analysis, internal flooding analysis, and thermal hydraulic analysis. To ensure that the IPE analytic models represented the as-built as-operated plant, analysts performed plant (system) walkdowns, interviewed key plant personnel, and performed simulator exercises. System work packages were developed which involved the review of plant system documentation and operational requirements and procedures. In addition and consistent with NUREG-1335 guidelines, the IPE underwent peer reviews by both in-house management and operations personnel, and outside consultants.

The FitzPatrick IPE did not identify any vulnerabilities requiring immediate action or strategies. However, a number of actions, including plant modifications, are under evaluation that would reduce the risk of core damage and loss of containment function. Implementation of the plant modifications, however, does not affect the overall conclusions of the IPE as the absolute risk reduction from these plant modifications is not significant.

As a result of concerns raised in a NRC Diagnostic Evaluation Team Report dated December 3, 1991, and to obtain a better understanding of PASNY's IPE process, a decision was made to perform a more detailed "step 2" review. On January 27-29, 1993, the NRC IPE review team and contractors performed a site visit and walkthrough of plant areas important from a PRA perspective. Plant personnel and analysts involved in the technical analysis were interviewed, "tier 2" information (selected fault trees, notebooks, and associated calculations) audited, and the training simulator visited.

Based on the review of the FitzPatrick IPE submittal and associated documentation, the NRC staff concludes that the licensee has met the intent of GL 88-20. This conclusion is based on the following findings: (1) the IPE is complete with respect to the information requested in GL 88-20 and associated NUREG-1335 submittal guidance document; (2) the front-end systems analysis, the back-end containment performance analysis, and the human reliability analysis are technically sound and capable of identifying plant-specific vulnerabilities to severe accidents; (3) PASNY employed a viable means (walkdowns) to verify that the IPE reflected the current plant design and operation; (4) the PSA which formed the basis of the IPE had an extensive peer review; (5) PASNY participated fully in the IPE process consistent with the intent of GL 88-20; (6) PASNY appropriately evaluated FitzPatrick's decay heat removal (DHR) function for vulnerabilities, consistent with the intent of the USI A-45 resolution; and, (7) PASNY responded appropriately to recommendations stemming from the containment performance improvement (CPI) program. In addition, PASNY is actively utilizing the IPE as a living document to enhance plant safety.

The NRC staff finds PASNY's approach to evaluating vulnerabilities appropriate, and the conclusion reasonable that no fundamental weakness or severe accident vulnerabilities now exist at FitzPatrick. The staff finds the FitzPatrick IPE process capable of identifying severe accident risk contributors or vulnerabilities, and that such capability is consistent with the objective of GL 88-20.

Enclosure 1 is the NRC Staff Evaluation of the FitzPatrick IPE. Enclosure 2 is the Technical Evaluation Report (TER) for the front-end analysis. Enclosure 3 is the TER for the back-end analysis. Enclosure 4 is the TER for the human reliability analysis.

This concludes the NRC staff review efforts associated with TAC No. M74411.

Sincerely,

ORIGINAL SIGNED BY:

Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. NRC Staff Evaluation of FitzPatrick IPE
2. TER for front-end analysis
3. TER for back-end analysis
4. TER for human reliability analysis

cc w/enclosures:
See next page

Distribution:

Docket File	CVogan	LReyes, 12/G/18
NRC & Local PDRs	BMcCabe	RHernan, 14/C/7
PDI-1 Reading	OGC	JFlack, NLS324
SVarga	ACRS (10)	MVirgilio, 8/E/2
JCalvo	CCowgill, RGN-I	JJohnson, 16/G/15
RACapra		

LA:PDI-1	PM:PDI-1 BCM	D:PDI-1 Rcw			
CVogan cw	BMcCabe:smm	RACapra			
5/2/94	5/3/94	5/9/94	/ /	/ /	/ /

OFFICIAL RECORD COPY
FILENAME: FIT74411.LTR

the USI A-45 resolution; and, (7) PASNY responded appropriately to recommendations stemming from the containment performance improvement (CPI) program. In addition, PASNY is actively utilizing the IPE as a living document to enhance plant safety.

The NRC staff finds PASNY's approach to evaluating vulnerabilities appropriate, and the conclusion reasonable that no fundamental weakness or severe accident vulnerabilities now exist at FitzPatrick. The staff finds the FitzPatrick IPE process capable of identifying severe accident risk contributors or vulnerabilities, and that such capability is consistent with the objective of GL 88-20.

Enclosure 1 is the NRC Staff Evaluation of the FitzPatrick IPE. Enclosure 2 is the Technical Evaluation Report (TER) for the front-end analysis. Enclosure 3 is the TER for the back-end analysis. Enclosure 4 is the TER for the human reliability analysis.

This concludes the NRC staff review efforts associated with TAC No. M74411.

Sincerely,

Robert A. Capra

Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. NRC Staff Evaluation of FitzPatrick IPE
2. TER for front-end analysis
3. TER for back-end analysis
4. TER for human reliability analysis

cc w/enclosures:
See next page

cc:

Mr. Gerald C. Goldstein
Assistant General Counsel
Power Authority of the State
of New York
1633 Broadway
New York, New York 10019

Resident Inspector's Office
U. S. Nuclear Regulatory Commission
P.O. Box 136
Lycoming, New York 13093

Mr. Harry P. Salmon, Jr.
Resident Manager
James A. FitzPatrick Nuclear
Power Plant
P.O. Box 41
Lycoming, New York 13093

Mr. J. A. Gray, Jr.
Director Nuclear Licensing - BWR
Power Authority of the State
of New York
123 Main Street
White Plains, New York 10601

Supervisor
Town of Scriba
Route 8, Box 382
Oswego, New York 13126

Mr. S. David Freeman, President
and Chief Executive Officer
Power Authority of the State
of New York
123 Main Street
White Plains, New York 10601

Charles Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, New York 10271

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

Ms. Donna Ross
New York State Energy Office
2 Empire State Plaza
16th Floor
Albany, New York 12223

Mr. Leslie M. Hill
Vice President - Appraisal
and Compliance Services
Power Authority of the State
of New York
123 Main Street
White Plains, New York 10601

ENCLOSURE 1

STAFF EVALUATION OF THE FITZPATRICK INDIVIDUAL PLANT EXAMINATION

(IPE)

(INTERNAL EVENTS ONLY)

TABLE OF CONTENTS

EXECUTIVE SUMMARY

The NRC staff completed its review of the internal events portion of the FitzPatrick Individual Plant Examination (IPE) submittal and associated information. The latter included licensee responses to staff generated questions seeking clarification of the licensee's process, audit of "tier 2" information held at the licensee site, plant walkdowns and interviews with key personnel involved in the IPE process.

The licensee's IPE is based on a FitzPatrick level 1 Probabilistic Risk Assessment (PRA) and a containment performance analysis consistent with Generic Letter (GL) 88-20 Appendix 1 guidelines. NUREG-1150 (Peach Bottom) insights and methodology were utilized extensively, and differences between the plants were accounted for in the models. The Power Authority of the State of New York (PASNY) personnel maintained involvement throughout the development and application of probabilistic risk assessment techniques to the FitzPatrick facility, with the objective of bringing PSA technology in-house. The staff notes that major plant departments contributed to the IPE/PRA development. Science Application International Corporation (SAIC) and Risk Management Associates provided technical support, primarily as reviewers and by contributing expertise in specific areas, such as human failure data analysis, common cause data analysis, internal flooding analysis, and the thermal hydraulic analysis.

The IPE estimated the overall core damage frequency (CDF) for the FitzPatrick plant to be 1.9 E-6/yr, a factor of about two less than that calculated in NUREG-1150 for a similar design (Peach Bottom). Differences primarily stem from anticipated transient without scram (ATWS) sequences which were substantially less significant in the FitzPatrick analysis than in Peach Bottom. (ATWS contributed 42% to the total core damage frequency at Peach Bottom, but less than 1% to FitzPatrick.) Loss-of-coolant accidents (LOCAs) were also less significant in the FitzPatrick analysis than in Peach Bottom due to differences in system models and assumed operator actions.

The FitzPatrick IPE identified station blackout (SBO) as the single largest contributor (91.1%), followed by transients with stuck-open safety relief valves (SRV)(s) and loss of all emergency core cooling system (ECCS) injection (6.2%), and transients with loss of containment heat removal (1.6%). Although loss of containment heat removal is a small contributor to overall CDF, 90% stems from loss of one of two vital safeguards buses. Vital bus failure at FitzPatrick is significant as it disables 3 of 4 decay heat removal paths (unlike, for example, Peach Bottom which disables only 1 of 4). Containment venting also plays a major role at FitzPatrick by reducing the overall CDF by a factor of 14.

The staff's review of the IPE's plant-specific hardware failure and unavailability data indicated that the licensee's analysis considered only 6 years of operation (8/80-9/86). Because of more recent experiences at FitzPatrick, the licensee has stated that they intend to update their component database as part of their "living" PRA program.

The licensee's IPE submittal did not identify any vulnerabilities requiring immediate action or strategies. However, a number of actions are under evaluation that would reduce the risk of core damage and loss of containment function. Implementation of the plant modifications, however, does not affect the overall conclusions of the IPE as the absolute risk reduction from these plant modifications is not significant.

Based on the review of the FitzPatrick IPE submittal and associated documentation, the staff concludes that the licensee has met the intent of GL 88-20. This conclusion is based on the following findings: (1) the IPE is complete with respect to the information requested in GL 88-20 and associated NUREG-1335 submittal guidance document; (2) the front-end systems analysis, the back-end containment performance analysis, and the human reliability analysis are technically sound and capable of identifying plant-specific vulnerabilities to severe accidents; (3) the licensee employed a viable means (walkdowns) to verify that the IPE reflected the current plant design and operation; (4) the PSA which formed the basis of the IPE had an extensive peer review; (5) the licensee participated fully in the IPE process consistent with the intent of GL 88-20; (6) the licensee appropriately evaluated FitzPatrick's decay heat removal (DHR) function for vulnerabilities, consistent with the intent of the USI A-45 resolution; and (7) the licensee responded appropriately to recommendations stemming from the containment performance improvement (CPI) program. In addition, the licensee is actively utilizing the IPE as a living document to enhance plant safety.

It should be noted that the staff's review primarily focused on the licensee's ability to examine FitzPatrick for severe accident vulnerabilities. Although certain aspects of the IPE were explored in more detail than others, the review is not intended to validate the accuracy of the licensee's detailed findings (or quantification estimates) which stemmed from the examination.

I. BACKGROUND

On November 23, 1988, the NRC issued GL 88-20 which requires licensees to conduct an IPE in order to identify potential severe accident vulnerabilities at their plant, and report the results to the Commission. Through the examination process, a licensee is expected to: (1) develop an overall appreciation of severe accident behavior, (2) understand the most likely severe accident sequences that could occur at its plant, (3) gain a more quantitative understanding of the overall probabilities of core damage and fission product releases, and (4) if necessary, reduce the overall probability of core damage and radioactive material releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

As stated in Appendix D of the IPE submittal guidance document NUREG-1335, all IPEs are to be reviewed by NRC teams to determine the extent to which each licensee's IPE process met the intent of GL 88-20. The IPE review itself is a two-step process; the first step, or "step 1" review, focuses on completeness and the quality of the submittal. Only selected IPE submittals, determined on a case-by-case basis, will be investigated in more detail under a second step or "step 2" review. The decision to go to a "step 2" review is primarily based on the ability of the licensee's methodology to identify vulnerabilities, and the consistency of the licensee's IPE findings and conclusions with previous PSA experience. A unique design may also warrant a "step 2" review to better understand the implication of certain IPE findings and conclusions.

On September 13, 1991, PASNY submitted the FitzPatrick IPE in response to GL 88-20 and associated supplements. The IPE submittal is based on an internal events level 1 PRA, and a level 2 containment performance assessment consistent with the guidance provided in GL 88-20 Appendix 1. The IPE process also addressed internal flooding. The licensee plans to provide a separate submittal for the individual plant examination of externally initiated events (IPEEE), which will be reviewed separately within the framework prescribed in GL 88-20 Supplement 4.

On February 6, 1992, and May 20, 1992, the staff forwarded questions to the licensee seeking additional information and clarification. The licensee responded in letters dated May 28, 1992 and September 1, 1992. As a result of concerns raised in a NRC Diagnostic Evaluation Team Report dated December 3, 1991, and to obtain a better understanding of the licensee's IPE process, a decision was made to perform a more detailed "step 2" review. The staff initiated contracts with Science & Engineering Associates, Inc. to audit level 1 system models; Scientech Inc. and Energy Research Inc. to audit level 2 accident progression and containment performance models; and Concord Associates to audit human reliability models. On January 27-29, 1993, the review team and contractors performed a site visit and walkthrough of plant areas important from a PRA perspective. Plant personnel and analysts involved in the technical analysis were interviewed, "tier 2" information (selected fault trees, notebooks, and associated calculations) audited, and the training simulator visited. The contractors' reviews are documented in the following Technical Evaluation Reports (TERs):

James A. FitzPatrick Step-2 IPE: Front End Audit [SEA 93-553-05-A:1];

Step 2 Review J.A. FitzPatrick Nuclear Plant IPE Submittal Human Reliability Analysis [CA/TR-93-19-05];

Technical Evaluation Report of the J.A. FitzPatrick Individual Plant Examination (IPE) Back-end Submittal [ERI/NRC 93-102].

These reports document findings and conclusions which stemmed from the NRC review of the IPE submittal and associated information including responses to staff questions, and information audited at the site, and contractor TERs. Specific numerical results and other insights taken from the licensee's IPE submittal are listed in the appendix.

II. STAFF'S REVIEW

1. Licensee's IPE Process

PASNY personnel maintained involvement throughout the development and application of PRA techniques to the FitzPatrick facility, with the objective of bringing PSA technology in-house. The staff notes that major plant departments provided input to the IPE/PRA development. Outside consultants, Science Application International Corporation (SAIC) and Risk Management Associates, provided technical support, primarily by providing expertise in specific areas, e.g., human failure data analysis, common cause data analysis, internal flooding analysis, thermal hydraulic analysis. To ensure that the IPE analytic models represented the as-built as-operated plant, the licensee analysts performed plant (system) walkdowns, interviewed key plant personnel, and performed simulator exercises. System work packages were developed which involved the review of plant system documentation and operational requirements and procedures. In addition and consistent with NUREG-1335 guidelines, the IPE underwent peer reviews by both in-house management and operations personnel, and outside consultants.

The IPE submittal documents and describes the techniques used to address each of the three major technical areas: the level 1 (front-end) systems analysis, level 2 (back-end) containment performance analysis, and the human reliability analysis. The methodology chosen for performing the FitzPatrick IPE analysis is consistent with the methods of examination identified in GL 88-20 and included a level 1 PRA, and a level 2 containment performance analysis that capitalized and utilized many of the NUREG-1150 Peach Bottom results.

The licensee defined core damage as a plant condition when the reactor water level is less than 2 feet above the bottom of the active fuel. The licensee has taken this definition and coupled it with a small event tree, linked large fault tree methodology to perform core damage analysis. By using this methodology, the licensee was able to identify dominant contributors expressed in terms of accident sequences, individual components, common cause failures, and human errors. Lists of dominant event contributors to three importance measure categories were generated, specifically risk reduction, risk increase, and uncertainty. The importance measures were used to evaluate "vulnerabilities," defined as those events that contribute most to risk increases (if their probability increases), risk reduction (if their probability decreases), and uncertainty. By reviewing the analytic results, the licensee identified potential plant vulnerabilities and associated safety enhancements. The IPE submittal, however, did not identify any vulnerabilities requiring immediate action or strategies.

Based on the review of the FitzPatrick IPE process, the staff finds the licensee's approach to evaluating vulnerabilities appropriate, and the conclusion reasonable that no fundamental weakness or severe accident vulnerabilities now exist at FitzPatrick. The staff finds the FitzPatrick IPE process capable of identifying severe accident risk contributors or vulnerabilities, and that such capability is consistent with the objective of GL 88-20.

2. Front-End Analysis

The staff examined the licensee's front-end systems analysis for completeness and consistency with accepted PRA practices. The IPE utilized the small event tree/large fault tree PRA methodology, consistent with methods identified in GL 88-20 for performing the IPE. The analysis capitalized on NUREG-1150 insights and industry performed PSAs. As part of the IPE process, the licensee implemented a PC-based version of the integrated plant model which is expected to be exercised and updated as part of a "living" PRA program.

The licensee's IPE process identified and modelled both generic and plant-specific initiators (including internal flood), and dependencies that exist between initiating events and the associated mitigating systems. Initiating events were found to be consistent with those identified in previous PRAs, including PRAs performed by the NRC staff. Functional event trees were developed for each initiator group. Special initiators were treated separately and included loss of safeguard AC buses, and loss of 125 VDC battery control boards. The IPE utilized the final safety analysis report (FSAR) success criteria and core cooling information developed as part of the NUREG-1150 risk analyses in establishing plant-specific success criteria for each initiator group. The analysis also included information developed as part of the staff's resolution of USI A-47 (NUREG/CR-1217), for scenarios involving the overfilling and overcooling of the reactor vessel. The IPE systems analysis addressed all front-line and support systems important to the prevention and mitigation of core damage accidents, including dependencies within plant systems and between systems, i.e., frontline systems-to-support systems, and support systems-to-support systems. Detailed dependency tables were developed and provided in the IPE submittal.

Selected portions of fault trees were audited during the site visit. These included:

a) Reactor Protection System
b) Alternate Rod Insertion System
c) Emergency Service Water System
d) Emergency Diesel Generator System
e) Reactor Core Isolation Coolant (RCIC) Enclosure Ventilation System
f) Residual Heat Removal/Low-Pressure Coolant Injection (LPCI) System
g) Control Rod Drive (CRD) System (coolant injection function)
h) 125 VDC System
i) Offsite Power

In general, audited fault trees were found to contain sufficient detail, with components and associated failure modes appropriately modeled. The supporting fault tree "work packages," which included systems notebooks, were found to be extensive and complete. The audit investigated the treatment of "logic loop" dependencies, (e.g., between the diesel generators and the emergency service water system), and found the loop properly broken, i.e., failure modes were not lost in the modeling process.

The submittal documented the quantification process and methods used to treat data analysis. Generic data sources included accident sequence evaluation program (ASEP) data sources listed in NUREG/CR-4550, data developed for the NRC Risk Method Integration Evaluation Program, and other previously performed PRAs. Plant-specific data was incorporated into the model by utilizing Bayesian techniques to update generic data. The staff notes, however, that plant-specific hardware failure and unavailability data represented only 6 years of operation (8/80-9/86). Because of more recent experiences at FitzPatrick, the licensee has stated that they intend to update their component database as part of their "living" PRA program. This should result in a better understanding of the impact of plant-specific data on the susceptibility of the FitzPatrick unit to severe accidents, and may identify where potential improvements can be made.

The licensee utilized the beta factor method described in NUREG/CR-4550 and generic data for treating common cause failure. During the audit, however, the staff noted that the analysis did not explicitly treat common cause failure of the emergency service water pump discharge check valves, and diesel generator ventilation fan and dampers. Subsequent sensitivity studies performed by the licensee indicated that adding the failures would not change the overall conclusions of the IPE. The staff, however, believes that the licensee would benefit from further consideration of these potential failures within their ongoing failure trending program.

The internal flood analysis used probabilistic and deterministic judgments in conjunction with three major elements:

1. the identification of potential flood areas and flood zones,
2. the identification of flooding scenarios and initial elimination of unimportant scenarios,
3. the quantification of remaining potentially important flooding scenarios.

The plant examination included flood walkdowns and consideration of flood-induced failure modes such as spraying and splashing. Train separation and redundancy requirements at FitzPatrick substantially reduced the significance of flooding and associated contribution to the overall core damage frequency estimate. The site visit focused on potential flooding scenarios that could fail equipment in separated areas, i.e., included a walkthrough of various reactor building levels and crescent areas, review of plant diagrams to assess the potential for backflow via equipment and floor drains. The audit did not identify any deficiencies.

The IPE estimated the overall core damage frequency for the FitzPatrick plant to be 1.9 E-6/yr, a factor of about two less than that calculated for Peach Bottom in NUREG-1150. Differences primarily stem from failure to scram sequences, which were found to be substantially less significant for the FitzPatrick plant than for Peach Bottom. (ATWS contributed 42% to the total core damage frequency at Peach Bottom, but less than 1% to FitzPatrick.) Unlike the NUREG-1150 analysis, the FitzPatrick IPE credited alternate boron injection capability, and estimated a lower human error probability for actuation of the standby liquid control system. RCIC had also been credited for maintaining water level and boron mixing in the FitzPatrick IPE, and procedures had been implemented to override the isolation logic of the main steam isolation valves (MSIVs) which were assumed closed in the Peach Bottom analysis.

LOCAs were also found to be less significant for FitzPatrick than for Peach Bottom. Differences primarily stem from credit taken for the control rod drive (CRD) system injection during medium LOCA conditions, and operator action involving manual opening of injection valves in conjunction with the use of the condensate system for large LOCA.

The single largest contributor identified in the FitzPatrick IPE is SBO (91.1%), followed by transients with stuck-open SRV(s) and loss of all ECCS injection (6.2%), and transients with loss of containment heat removal involving loss of 4.16-kv safety bus (1.6%). Although SBO for the FitzPatrick plant is high relative to other contributors, the absolute CDF contribution is small due to the four emergency diesel generators and plant location which is away from the eastern coastline. Long-term SBO sequences nevertheless dominate, with loss of coolant injection upon battery depletion being the largest contributor to the blackout sequences. For the short-term blackout sequences, random failure of the batteries dominate at FitzPatrick because the configuration maintains only two batteries and associated control boards.

The staff finds the licensee's front-end IPE analysis complete, with documentation consistent with the information requested in NUREG-1335. The employed analytical techniques were found to be consistent with other NRC reviewed and accepted PSAs. The licensee identified and expanded the most probable core damage sequences to identify dominant contributors, i.e., specific components, plant conditions or behavior, or common cause failures that contribute to plant vulnerabilities. Importance measures were generated to aid in the evaluation of the dominant contributors. The staff, therefore, finds the FitzPatrick IPE front-end analysis meets the intent of GL 88-20.

3. Back-End Analysis

The staff examined the licensee's back-end analysis for completeness and consistency with the guidance specified in GL 88-20, Appendix 1. During the site audit, the review team interviewed licensee analysts and performed walkthroughs of the crescent room between the reactor building and containment, and areas containing containment vent valves, vent path, and connections to the standby gas treatment system (SGTS). Information specifically audited at the site include:

- Containment Performance Improvement
  - EOP-4 "Primary Containment Control"
  - AOP-35, Revision 8 "Post Accident Venting of the Primary Containment"
- Containment Event Trees (CETs) and quantification
- Containment capacity and failure characterization
- Secondary building hydrogen combustion issues
- Pressure and temperature histories and consistency of computer codes used for the analyses.

The FitzPatrick back-end IPE containment performance and source term analysis utilized the NUREG-1150 Peach Bottom methodology. A comparison of Peach Bottom and FitzPatrick Plant containment design features that contribute to the progression of severe accidents is contained in the contractor's back-end TER. The level 2 approach involved key phenomena and processes that could occur during the evolution of severe accidents, and which could subsequently impact containment and containment system performance. Accident progression codes included Rev. 1.6 of the BWRSAR Code to treat in-vessel melt progression, including core debris relocation and release from the reactor vessel after vessel failure; the EVNTRE Code to process containment event trees; and version 1.7 of the HECTR code to address hydrogen combustion outside the primary containment.

The methodology specifically included the interface between the level 1 front-end systems analysis and level 2 containment performance analysis. Level 1 sequences having a similar effect on plant performance were binned into groups, or plant damage states representing the status of core cooling systems and containment systems at the time of core damage. These states were systematically analyzed by CETs which provide a structured approach for assessing containment phenomena and accident progression. (The IPE utilized the Peach Bottom NUREG-1150 CETs which contains over 145 questions, 19 of which were modified to account for FitzPatrick plant-specific features.) The analysis also considered containment isolation failure and the potential susceptibility of penetration elastomer seal material to prolonged high temperature. All accident sequences (represented by plant damage states) that met GL 88-20, Appendix 2 screening criteria were analyzed by utilizing the CET methodology.

The IPE characterized the containment performance for each of the CET end-states by assessing containment loading. The licensee referenced the Peach Bottom NUREG-1150 structural analysis and performed a comparison analysis with respect to the construction material and major structural components in the drywell and torus to determine containment failure pressure. The analysis indicated that a 12% reduction in the Peach Bottom containment capacity was warranted for the FitzPatrick design, primarily because of differences in thickness of the top part of the torus shell and vent line bellows. The CET accident progression sequences were consolidated into release bins, each of which had an assigned fission product release characteristic (NUREG-1150 source terms).

The CETs were quantified, and the most probable containment failure mechanisms identified. Consistent with NUREG-1150, the analysis indicated that drywell liner attack dominated containment failure at FitzPatrick. Small differences in late containment failure probability between the FitzPatrick IPE and Peach Bottom were generally associated with differences in estimated containment failure pressures, i.e., 140 psig (FitzPatrick) vs 150 psig (Peach Bottom). Sensitivity studies were performed to better understand the impact of phenomenological uncertainties and recovery actions. Containment failure as a percentage of total CDF comparison to Peach Bottom NUREG-1150 are provided in the table below. (Note: the FitzPatrick IPE considers containment venting as failure).

The IPE's use of the Peach Bottom CET, and associated split fractions from NUREG-1150 with regard to drywell liner failure resulted in a high (53.6%) probability of early drywell failure. More recent studies (NUREG/CR-5423), however, indicate that for cases in which water is on the drywell floor, and which can be replenished ("wet" case), liner failure is very unlikely with failure probabilities estimated in the 10^{-4} to 10^{-6} range. The licensee studies, however, indicate that because of plant-specific features (equipment sumps inside the reactor pedestal region), the potential benefit of the "wet" case could not be credited. (Based on the FitzPatrick design, corium on the drywell floor would likely lead to base mat oblation and failure of the drywell shell below the equipment sumps inside the reactor pedestal). The licensee maintains that the higher likelihood of liner failure in the IPE analysis is appropriate given the level of uncertainty surrounding the issue.

Containment Failure	Peach Bottom/ NUREG-1150	FitzPatrick IPE
CDF (per year)	Internal $4.5x10^{-6}$	Internal $1.9x10^{-6}$
Early/Drywell Failure	52.4	53.6
Early/Wetwell Failure	3.3	6.8
Late/Drywell Failure	4.7	11.6
Late/Wetwell Failure	0.3	14.4
Wetwell Venting	11.0	na
Intact	28.0	13.6

The staff, nevertheless, believes (based on the more recent studies) that the "wet" case will reduce the likelihood of liner melt-through, in addition to providing fission product scrubbing. The licensee, therefore, could benefit from investigating the issue further as part of the follow-on accident management program, to remove any conservatism from the analysis that could potentially mask a beneficial accident management strategy. The staff's review did not identify any obvious or significant problems or errors in the back-end analysis. The overall assessment of the back-end analysis is that the licensee has made reasonable use of PSA techniques in performing the back-end analysis, and that the techniques employed were capable of identifying severe accident vulnerabilities. The staff, therefore, finds the FitzPatrick IPE back-end analysis meets the intent of GL 88-20.

4. Human Factor Considerations

The Human Reliability Analysis (HRA) portion of the IPE is based primarily on the Accident Sequence Evaluation Program - Human Reliability Analysis Procedure (ASEP-HRAP) described in NUREG/CR-4772. In addition to ASEP-HRAP, elements of the Systematic Human Action Reliability Procedure (SHARP) were used in the representation of complex diagnosis events. The licensee identified and modeled two types of human events, those activities that may disable a system (i.e., pre-accident human events) and those activities needed to mitigate an accident (i.e., post-accident human events). The human events were modeled in the event trees as a top event and in the fault trees as a basic event.

Pre-accident human actions modeled in the IPE are events that were identified by gathering plant-specific information from FitzPatrick surveillance, calibration and maintenance procedures, from scram reports, and from Licensee Event Reports. To identify the more critical pre-accident human events, the ASEP-HRAP guidelines were used; no quantitative screening analysis was employed for these events. The staff's examination of the pre-accident event identification process verified that administrative controls exist to assure appropriate restoration of equipment after an activity is completed; that appropriate plant personnel (from operations, maintenance and I&C) participated in identifying pre-accident events and reviewed the results; and that a select number of diverse functional test/calibration activities were observed by the licensee's HRA analysts to confirm the capability of their process to identify potential human errors.

Post-accident human actions modeled in the IPE are operator actions dictated by the Emergency Operating Procedures (EOPs) (usually modeled in the event trees) or operator actions to recover a failed system (recovery actions) identified from the Abnormal Operating Procedures, Operating Procedures and simulator observations. An iterative quantitative process was used to identify the most critical post-accident human actions. Review of specific examples of these screening processes did not identify any areas of concern.

To derive human error probabilities, basic human error probability data were obtained from NUREG/CR-4772 and were modified by performance shaping factors to account for influences on operator performance. Performance shaping factors accounted for dynamic vs step-by-step actions, for stress levels, for number of control room operators, and action complexity. They were developed by using plant-specific information and the ASEP-HRAP method. In addition, considerable credit was taken for the use of the symptom-based EOPs and the quality of FitzPatrick procedures and training, by applying adjustment factors to the basic human error probabilities.

The staff examined whether the adjustments were supported by detailed evaluation of the performance shaping factors which influence operator behavior and whether dependencies were considered and accounted for in the analysis. The review results supported the IPE's post-accident human error treatment with due consideration of operator training and demonstrated performance in routine requalification simulator training. Twenty-two accident scenarios were run on the simulator in support of the HRA. Different operating crews participated in the sessions, and shift staffing levels were consistent with Technical Specification requirements. Furthermore, the review confirmed that dependencies were treated appropriately.

As stated in the IPE, four human recovery events reduced core damage frequency resulting from internal causes by a factor of 3.7. These events are: (1) initiation of standby liquid control during ATWS, (2) controlling reactor water level at the top of active fuel and using control rod drive system to inject boron should the standby liquid control (SLC) fail, (3) manual opening emergency core cooling system injection valves during transients that result from stuck-open SRVs and LOCA, should LPCI system fail, (4) enhancing CRD system flow to provide coolant in various transients. During the plant visit the staff reviewed the reasonableness of the licensee assumptions made about the accessibility of equipment, manual actions required, etc., and the rigor of the process applied in performance of the licensee's walk-downs. The staff concludes that the licensee's approach is reasonable.

In summary, the staff finds the licensee's assessment of human reliability as capable of discovering severe accident vulnerabilities from human errors and consistent with the intent of GL 88-20. The HRA methodology described in the licensee's submittal supports the quantitative understanding of the overall probability of core damage during plant operations, as well as, an understanding of the contribution of human actions to that probability. Human related plant improvements that are under review, such as preventing miscalibration errors by using different crews to calibrate different divisions and institute independent checking, are expected to enhance the human reliability and plant safety. In addition, the licensee's stated intention to maintain a "living IPE" will ensure that a mechanism exists to continue to identify and evaluate the risk significance of potentially important human actions during plant operation and maintenance.

5. Containment Performance Improvements (CPI)

In addition to the implementation of the hardened vent, GL 88-20, Supplement 1, contains CPI recommendations that are to be considered by licensees of Mark I plants during the development of their IPEs. These items include the following:

a) Alternate Water Supply for Drywell Spray/Vessel Injection,
b) Enhanced Reactor Pressure Vessel (RPV) Depressurization System Reliability, and
c) Emergency Procedures and Training (Revision 4 of the BWR Owners Group Emergency Procedure Guidelines).

The FitzPatrick IPE addressed containment venting as a means by which the conditional probability of containment failure (and subsequent core damage) can be reduced, in addition to supporting mitigation of severe accidents. Containment venting reduced the core damage frequency at FitzPatrick by an estimated factor of 14.

Containment venting procedures currently require hard pipe venting of the wetwell air space anytime the containment pressure exceeds 44 psig. The vent path at FitzPatrick utilizes piping from the containment to the inlet transition piece of the standby gas treatment system (SBGT) filter train. Because the transition piece is located outside the reactor building pressure boundary, failure of the transition piece upon containment venting is limited to the SBGT system. The survivability and accessibility of vital plant equipment is, therefore, not compromised by failure of the transition piece.

Wetwell venting will normally be initiated at the primary containment and purge (PCP) panel located in the relay room. For accident sequences in which motive power is unavailable to the valves, the operators are expected to locally hand-wheel the valves open. Venting of the containment is accomplished using AOP-35 "Post Accident Venting of the Primary Containment." This procedure instructs the operator to vent the containment regardless of the radiological consequences. The procedure (for which operators have been trained) is currently entered from EOP-4 "Primary Containment control" before the containment pressure exceeds 44 psig.

During the plant visit, the staff reviewed the modeling of wetwell venting in the IPE, examined AOP-35 and EOP-4 with plant operations personnel and walked through the process of implementing AOP-35 from both the PCP panel and locally at each valve. The staff concludes that the wetwell venting function is appropriately modeled in the licensee's IPE analysis.

The FitzPatrick unit has cross ties between the diesel driven fire pumps and Residual Heat Removal Service Water (RHRSW) "A" header. The RHRSW "A" header can be cross-tied to the LPCI A injection path and provide an alternate source of low pressure injection, and delay accident progression during SBO. Implementation of this cross-tie, however, has only limited impact on loss-of-injection induced core damage frequency, because system failure is dominated by failure of the low pressure ECCS injection valves to open.

The licensee found that manual alignment of the fire protection system (FPS) pumps to the discharge of the RHRSW "A" header could reduce the probability of core damage in TW sequences that result from RHRSW pump failure. Therefore, the licensee is currently considering modifying procedures and operator training in order to support this action.

As part of the IPE program, the licensee did not implement modifications which would allow the use of the FPS as an alternate water supply for drywell sprays. Although currently under evaluation, the licensee has not resolved questions surrounding the ability of the FPS pumps to provide adequate flow to the drywell spray headers. The licensee did perform sensitivity studies to determine the effects of full drywell spray capability, i.e., during periods of loss of AC power. The results indicated that the availability of the drywell sprays reduces the probability of containment failure, delays containment failure, shifts the location of failure from drywell to wetwell, and enhances fission product decontamination. Based on the potential benefit of the enhancement, the staff believes that the licensee should continue to investigate liner melt-through in their accident management program, and consider more recent research developments and findings.

The licensee has examined the benefit of providing a portable diesel generator to charge the DC batteries to enhance the reliability of the reactor pressure vessel (RPV) depressurization system, and thus the ability of the plant to cope with SBO. The results indicated that a reduction in CDF could be better achieved through other changes (e.g., use of a FPS cross-tie to the emergency service water (ESW) to provide EDG jacket cooling). The licensee also evaluated the feasibility of increasing nitrogen supply pressure to the SRVs, to sustain their operability during TW and SBO events, but decided that other changes to reduce the core damage frequency were more practical.

In addition to the consideration of the above Mark I safety enhancements, GL 88-20 Supplement 1 also encourages licensees to implement Revision 4 of the BWR Owners Group Emergency Procedures Guidelines (EPGs). Revision 4 to the

BWR Owners Group EPGs was implemented in June of 1990 and has been accounted for in the FitzPatrick IPE event and fault tree models.

Based on the review of the licensee's IPE process, the staff concludes that the licensee response to the CPI Program recommendations, which included searching for vulnerabilities associated with containment performance during severe accidents, is reasonable and consistent with the intent of GL 88-20 and associated Supplement 1.

6. Decay Heat Removal Evaluation

In accordance with the resolution of USI A-45 "Shutdown Decay Heat Removal Requirements," the licensee performed an examination of the FitzPatrick DHR function to identify potential vulnerabilities. The examination identified transients with loss of long term containment heat removal sequences (TW) as only a small percentage (<2%) of the total CDF at FitzPatrick. Loss of one of two vital safeguards buses, however, contributes to 90% of the TW sequences. (Failure of a single AC bus disables three of four DHR paths, unlike, for example, Peach Bottom where loss of a single AC bus causes loss of only one of four DHR paths.)

The low contribution of the TW sequences to the total CDF is primarily based on effective wetwell/drywell venting, and inhibiting automatic switchover of the HPCI suction from the condensate storage tank (CST) to the pool. The former operator actions (containment venting) had been found to have a significant impact on the CDF at FitzPatrick, as it dominates all three importance measures related to decay heat removal: risk increase, risk reduction, and uncertainty importance, even though containment venting is not available during SBO.

The IPE assessed major safety functions following transients and LOCA events, including:

(a) Recovery of the power conversion system (PCS) in sequences which progress to long term loss of containment heat removal. Procedures have been implemented that would allow operators to reopen the MSIVs and bypass valves and recover from reactor isolation.

(b) Impact of severe accidents on the integrity of the reactor primary system and the piping for the recirculation system and the SRV discharge system (including the vacuum breakers), piping support systems, and the seals of the recirculation pumps (including the isolation valves) and the RHR pumps.

(c) Available plant design and operational features for independent means of providing short-term and long-term coolant injection (both high and low pressure) to the reactor. The licensee has identified plant-specific decay heat removal scenarios involving injection failures and utilization of HPCI, RCIC or the CRD for long-term coolant makeup to the reactor. Procedures have been implemented that would provide long-term makeup to the CST through the utilization of the demineralized water storage and transfer system.

(d) Plant design and operational features, and identified training requirements (such as inhibiting and overriding certain design-intended functions) for independent means of providing containment heat removal. Containment heat removal function includes the suppression pool cooling and the spray function through the RHR system, the shutdown cooling function through the RHR system, and the containment overpressure protection function through manual venting of the drywell and the wetwell. In situations where the RHR system is unavailable, the licensee has established containment venting procedures and associated operator training. Depressurization and use of the diesel-driven fire water pump is also an option during containment venting, which can provide low-pressure coolant makeup to the reactor to prevent a core damage event.

In accordance with the resolution of USI A-45, the licensee performed an examination of FitzPatrick to identify plant-specific DHR vulnerabilities, and potential design and procedure change options to improve DHR reliability. Based on that process, the staff finds the licensee's DHR evaluation consistent with the intent of GL 88-20, and resolution of USI A-45.

7. Licensee Actions and Commitments From the IPE

The licensee used the IPE process to identify plant and/or procedural modifications, and plans to maintain the PRA program "living." The licensee's long term involvement in the development of the FitzPatrick PRA, in addition to review of other PRAs, most notably the Peach Bottom NUREG-1150 PRA, has resulted in the incorporation of plant and/or procedural modifications prior to the IPE effort. These are all modeled in the latest FitzPatrick PRA and are an integral part of the IPE.

As a result of the IPE effort, certain improvements were specifically identified and implemented (or plan to be implemented) that would help reduce the likelihood of core damage and loss of containment heat removal. Improvements include:

a) increasing the RCIC turbine exhaust set points,

b) repowering the RCIC enclosure exhaust fans from AC to DC,

c) Fire Protection System modifications to provide EDG jacket water cooling directly or through the ESW system.

In addition, aligning FPS pumps to the drywell spray header is under consideration and dependent on FPS capability. Potential improvements and insights identified during the IPE process which have lead to recommendations and follow-on evaluations by the licensee include:

a) Provisions to prevent and/or mitigate the consequences of random failure or miscalibration of reactor pressure transmitters,

b) Increase of the N_2 supply pressure to SRVs to preclude loss of SRV operability due to increasing containment pressure,

c) Provide means to prevent HPCI failure resulting from HPCI suction auto-transfer to the torus,

d) Procedure change to instruct operators to remain above the HPCI/RCIC low-reactor trip points under certain scenarios,

e) Procedure change to use RCIC instead of HPCI to control reactor vessel water level in sequences involving a stuck open relief valve,

f) Limit reactor level to 118" above TAF (rather than 222.5") to preclude heat diversion to the torus for events with turbine by pass capability,

g) Modification of flow control valve to fail open upon loss of instrument air to allow one of the CRD flow paths available,

h) Provision for FPS Cross-tie and alignment of the FPS to allow for containment heat removal,

i) Provisions to readily access a portable diesel generator for recharging Class 1E batteries for SBO events.

j) Provision for operator action to provide an unlimited supply of water to the CRD.

k) Procedure revision to prevent HPCI and RCIC trip on low steam supply pressure caused by emergency depressurization,

l) Revision of AOP49 "SBO" to address bus recovery if safeguard busses are lost during a transient as a result of the failure of both safeguard bus tie breaker lockout relays, and

m) Protection of RCIC and HPCI motor control centers BMCCI and BMCC2 from spray and splash from internal flooding.

Although the review team did not examine the merits of the above recommendations in detail, the staff notes that the licensee is applying PRA/IPE findings to enhance plant safety consistent with the intent of GL 88-20. The staff, therefore, finds the licensee's actions and commitments reasonable for closure of severe accident concerns.

III. CONCLUSIONS

The staff finds the licensee's IPE submittal for internal events including internal flooding consistent with the information requested in NUREG-1335.

Based on the review of the submittal and associated information, including "tier 2" supporting information held at the site, the staff finds reasonable the IPE's conclusion that no fundamental weakness or severe accident vulnerabilities exist at FitzPatrick. The staff notes that:

(1) PASNY personnel were involved in the development and application of PSA techniques to the FitzPatrick facility, and that the associated walkdowns, personnel interviews, simulator exercises and documentation reviews constituted a viable process for confirming that the IPE represent the as-built, as-operated plant.

(2) The licensee's performed an in-house peer review to provide assurance that the IPE analytic techniques had been correctly applied and documentation was accurate.

(3) The front-end IPE analysis is complete with respect to the level of detail requested in NUREG-1335. In addition, the analytical techniques were found to be consistent with other NRC reviewed and accepted PSAs.

(4) The back-end analysis addressed the most important severe accident phenomena normally associated with Mark I containment types. No obvious or significant problems or errors were identified.

(5) The HRA allowed the licensee to develop an understanding of the contribution of human errors to CDF and containment failure probabilities.

(6) The employed analytical techniques in the front-end analysis, the back-end analysis, and the HRA are capable of identifying potential plant-specific vulnerabilities.

(7) The licensee's IPE process searched for DHR vulnerabilities consistent with the USI A-45 (Decay Heat Removal Reliability) resolution.

(8) The licensee responded to CPI Program recommendations, which include searching for vulnerabilities associated with containment performance during severe accidents.

Based on the above findings, the staff concludes that the licensee demonstrated an overall appreciation of severe accidents, has an understanding of the most likely severe accident sequences that could occur at the FitzPatrick facility, has gained a quantitative understanding of core damage and fission product release, and responded appropriately to safety improvement opportunities identified during the process. The staff, therefore, finds the FitzPatrick IPE process acceptable in meeting the intent of GL 88-20. The staff also notes that the licensee's intent to continue to use and maintain its PRA document will enhance plant safety and provide additional assurance that any potentially unrecognized vulnerabilities would be identified and evaluated during the lifetime of the plant.

APPENDIX
FITZPATRICK DATA SUMMARY SHEET*
(INTERNAL EVENTS)

- Total core damage frequency (CDF) : 1.92E-6/Year

- Contributions to dominant core damage sequences:

	Contribution
SBO (SBO)	91.1%
Transient with stuck-open SRV	6.2%
Transient with loss of containment heat removal (TW)	1.6%
ATWS	<1.0%
LOCAs	<1.0%

- Major operator actions to prevent core damage or containment failure:

  - Containment venting during loss of containment heat removal events.
  - Initiation of standby liquid control (SLC) during ATWS events
  - Controlling the reactor water level at the top of active fuel and using the control rod drive system to inject boron should the SLC system fail.
  - Manual opening of ECCS injection valves locally.
  - Enhancing CRD system flow to provide coolant in various transients.

- Conditional containment failure probability given core damage:
  (Note: containment venting considered as failure)

Early/Drywell Failure	53.6%
Early/Wetwell Failure	6.8%
Late/Drywell Failure	11.6%
Late/Wetwell Failure	14.4%
No Failure	13.6%

- Significant PRA findings:

  - The most significant risk-reduction events are:

    . Loss of offsite power initiator
    . Failure to recover offsite power in 13 hours
    . One stuck-open safety relief valve
    . ESW system loop B out for maintenance
    . Failure to recover offsite power in 5 hours

o The most significant risk-increase events are:

. Common cause failure of the batteries
. Common cause failure of the ESW pump to continue to run and to start on demand
. Mechanical failure of the reactor protection system
. Common cause failure of EDGs.

o Improvements stemming from IPE study:

. Increasing the RCIC turbine exhaust set points
. Repowering the RCIC enclosure exhaust fans from AC to DC
. FPS modifications to provide EDG jacket water cooling directly or through the ESW system.

o Important plant hardware and plant characteristics:

. Primary containment (drywell or torus) venting: hard piping
. Alternate boron injection: SLC-to-CRD pump
. FPS: cross-tie to RHRSW A
. RHR pump seal: cooling failure does not lead to pump failure
. Core spray pump seal: cooling failure does not lead to pump failure
. HPCI turbine: turbine exhaust trip at 150 psi
. MSIV isolation: low-level trip from 118 in. to 59.5 in.
. HPCI/RCIC high temperature trip: increased availability during SBO
. RCIC suction: no provision for auto transfer on high torus level
. EDGs: any one of four can provide shutdown

o Potential improvements under evaluation:

. Administrative changes to minimize reactor pressure transmitter miscalibration
. Increased nitrogen pressure for SRV
. Modify the HPCI logic on the auto transfer
. Limitation of a maximum reactor water level
. Procedural modification for CRD injection flow, and modification of the CRD flow control valve to fail safe or as-is on loss of instrument air
. Modification of procedures on fire protection system for containment heat removal.
. Providing portable generator for charging the 125 VDC batteries
. Procedural change on HPCI and RCIC trip: EOP-8
. Revise SBO AOP-49
. Providing protection of HPCI and RCIC from floods

(* Information has been taken from the FitzPatrick unit 1 IPE and has not been validated by the NRC staff.)

ENCLOSURE 2

FITZPATRICK INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(FRONT-END)