APPENDIX B

PILGRIM NUCLEAR POWER STATION INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(BACK-END)

ERI/NRC 96-102

TECHNICAL EVALUATION REPORT OF THE PILGRIM INDIVIDUAL PLANT EXAMINATION BACK-END SUBMITTAL

Final Report

May 1996

Energy Research, Inc.
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Prepared for:
SCIENTECH, Inc.
Rockville, Maryland

Under Contract NRC-04-91-068
With the United States Nuclear Regulatory Commission
Washington, D.C. 20555

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FINAL REPORT

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E. EXECUTIVE SUMMARY

This Technical Evaluation Report (TER) documents the findings from a review of the back-end portion of Pilgrim Nuclear Power Station Individual Plant Examination (IPE) Back-End submittal of the Boston Edison Company (BECo). The primary intent of the review is to ascertain whether or not, and to what extent, the back-end IPE submittal satisfies the major intent of Generic Letter (GL) 88-20 and achieves the four IPE sub-objectives. The review utilized both, the information provided in the IPE submittal, and additional information provided by the licensee in response to NRC questions.

The back-end portion of the IPE submittal supplies a substantial amount of information with regards to the subject areas identified in Generic Letter 88-20, and NUREG-1335.

E.1 Plant Characterization

The Pilgrim plant is a General Electric Company BWR/3 plant with a Mark I containment located at Plymouth, Massachusetts. The rated thermal power is 1998 MWt (687 MWe). The mean containment failure pressure is 98 psig.

E.2 Licensee's IPE Process

The IPE was a cooperative utility-contractor effort, with most of the work being performed by BECo staff. Tenera, Fauske and Associates, and Gabor, Kenton, and Associates were the contractors. Five BECo senior engineers were involved in the IPE. It appears that BECo staff were involved in many, if not all, aspects of the study. However, since no personnel breakdown by task is provided, it is impossible to ascertain the actual level of utility involvement. It is stated in Section 2.1.2.1 of the submittal, that BECo staff were involved in all areas of the IPE (both level 1 and level 2), and that a complete transfer of technology was accomplished. As stated in Section 2.1.2.2 of the submittal, the Pilgrim IPE underwent four levels of review. The first level of review involved a review of the IPE assumptions and results, principally performed by the BECo PRA/IPE staff, with limited support from consultants. The second level of review involved a review of other similar industry studies and plant-specific information from the Peach Bottom/NUREG-1150 study. An internal peer review provided the third level of review for the Pilgrim IPE. As stated in the submittal, "[t]his independent, in-house review was conducted to ensure the accuracy of the documentation contained in the report, and to validate the IPE process and results." The final level of review was performed by an external peer review team. Specifically in regards to the back-end analysis, it is stated in the submittal that an independent review was performed by the primary back-end contractor. While Section 2.1.2.2 of the submittal lists a number of important comments/insights that were obtained from the review process, however, these comments do not pertain to the back-end portion of the submittal.

The methodology employed in the Pilgrim submittal for the back-end evaluation is clearly described, and the IPE is logical, traceable, and consistent with GL 88-20. The definition of Plant Damage States (PDSs) involved using the core damage sequences as input to Containment

Safeguards Event Trees (CSETs). The CSET nodes represent those critical safety functions which describe the post core damage status of systems important to accident progression, containment response, and radiological release. Probabilistic quantification of severe accident progression involved development of a relatively small Containment Phenomena Event Tree (CPET), and use of supporting fault trees to tailor each CPET question to the specific PDS being evaluated. The results of the CPET analyses lead to an extensive number of end-states, which were in turn binned into a manageable number of release categories, based on similarities in accident progression and source term characteristics.

The front-end analyses in the IPE submittal report a Core Damage Frequency (CDF) of 5.8 x 10^{-5} per reactor year. The dominant contributors to core damage are high pressure core damage sequences (initiated by loss of offsite power or by transients such as loss of feedwater) with loss of coolant makeup (79.3%), followed by Anticipated Transient Without Scram (ATWS) sequences (7%), low pressure core damage sequences with loss of coolant makeup (6.2%), and LOCAs (3.2%).

After the IPE was submitted to the NRC, the licensee revised the models for CDF, including a number of enhancements. Specific examples include: elimination of the dependency of HPCI on room cooling, reduction in the number of Safety Relief Valves (SRVs) required for the success of the ADS, modification of DC power success criteria, development of recovery actions for SBO, common cause breaker failures, etc. In addition, revisions of the initial estimates for Loss of Offsite Power (LOOP) were also made in the IPE submittal. The net impact of these changes is the reduction of the CDF from 5.85 x 10^{-5} per reactor year to 2.84 x 10^{-5}. In addition, the relative contributions of the various PDSs to the CDF, were also found to be different than in the original submittal. However, the containment analyses were not revised.

The interface between the front-end (level 1) analysis and the back-end (level 2) analysis is accomplished through the propagation of front-end (core damage) sequences through the Containment System Event Trees (CSETs). The CSETs are used to define the status of systems which are important for analyzing containment response to accident challenges. Since the actual cutsets from the level 1 analysis are input to the CSETs, system dependencies are accounted for between the level 1 systems and the containment systems. The core damage/CSET sequences are grouped together into Plant Damage States (PDSs), based on functional characteristics important to accident progression, containment failure and source term definition. Only 12 of the PDSs actually contain cutsets.

Probabilistic quantification of severe accident progression for the probabilistically significant PDSs was performed using a Containment Phenomena Event Tree (CPET). The methodology employed in the Pilgrim submittal involved development of a relatively small CPET, and use of small, supporting fault trees to tailor each CPET question to the specific PDS being evaluated. The Pilgrim CPET contains the following nine nodes:

- Debris Cooled In-Vessel
- Small Lower RPV Head Failure

- No Pedestal Failure at RPV Failure
- No Drywell Failure at RPV Failure
- Debris Cooled Ex-Vessel
- No Drywell Over-Temperature Failure
- Containment Not Challenged By Over-Pressurization
- Wetwell Vapor Space Failure or Venting
- No Pool Bypass

A phenomenological fault tree was developed to support the quantification of each CPET heading listed above. The quantification of the fault tree basic events was obtained through one or more of the following sources:

1. Pilgrim-specific MAAP calculations,
2. Peach Bottom or Grand Gulf/NUREG-1150 supporting documentation [3,4], or
3. Pilgrim-specific "hand calculations."

The Pilgrim CPET includes most of the relevant phenomena for BWRs with Mark I containments; nevertheless, as discussed in this TER, the quantification of their impact on the Pilgrim containment is weak.

The results of the CPET analyses lead to an extensive number of end-states, which are classified into a manageable number of release categories, categorized by similarities in accident progression and source term characteristics. The sequence characteristics which were identified in the IPE submittal as having the greatest impact on fission product release at Pilgrim are the following:

- Containment Bypass
- Debris Cooled In-Vessel
- Time of Containment Failure/Venting (Relative to Core Damage)
- Mode/Location of Containment Failure (or Venting)
- Suppression Pool Bypass
- Type of Ex-Vessel Core/Concrete Interactions (Dry, Wet or None)

Using these characteristics as headings, a Source Term Category Grouping Diagram was developed. The end points of this diagram represent the individual release categories. A total of 34 release categories are defined by the submittal, of which only 20 are reported as non-zero (i.e., frequency > 10^{-10}/ry). MAAP calculations were performed to determine the source terms for representative sequences of 12 of the 20 non-zero release categories. Of the remaining eight release categories, the source terms for seven were determined to be similar to one of the previously calculated source terms. The final release category involved no containment failure or venting, and was assigned no source term. A summary of the releases associated with each non-zero release category is provided in Table 4.7-5 of the submittal. The source terms reported in this table do not take credit for any fission product retention in the reactor building.

E.3 Back-End Analysis

The conditional probabilities of early and late containment failure calculated by the submittal are 0.216 and 0.61, respectively (see Table E.1). The conditional probability of intact containment is about 0.17.

From review of Table E.1, it is seen that the major difference between Pilgrim and the Peach Bottom plant (NUREG-1150 analyses) in the table, is that Pilgrim has a lower conditional probability of early containment failure, and accordingly, a higher probability of late containment failure. It appears that Pilgrim has understated the contribution from early drywell failure due to overpressurization (see Section 2.1.3.3 of this review). Also, for BWRs with Mark I containments, the dominant cause of early containment failure is drywell liner melt through. The submittal (p. 4.7-10) states that the principal reason for the difference in early containment failure is due to the Pilgrim assumption that for sequences with high vessel pressures at RPV failure, the probability of a liner melt through in a dry cavity is only 0.1, if in-vessel injection is available following RPV failure. Also, as discussed in Section 2.1.2.2 of this review, the contribution of drywell liner melt through reported in the submittal, is somewhat lower due to the incorporation of the results from a more recent study. However, since the submittal does not provide the fraction of core damage frequency that is associated with a dry cavity at the time of RPV failure, as well as other event probabilities associated with the probability of liner melt through, it is not possible to determine the exact reasons why Pilgrim exhibits a much lower probability of early containment failure.

The relatively high probability of late containment failure reported in the submittal may be partially the result of the Pilgrim containment flooding strategy, which directs the operators to flood the containment if RPV water level cannot be maintained above the top of active fuel for non-ATWS sequences, or above two-thirds core height for ATWS sequences. Once containment

Table E.1 Containment Failure as a Percentage of Internal Events CDF: Comparison of Pilgrim IPE Results to Peach Bottom NUREG-1150 Results

Containment Failure	Peach Bottom NUREG-1150	Pilgrim IPE
CDF (per year)	4.3×10^{-6}	2.84×10^{-5}
Early Failure	46	21.6
Bypass	-	0.4
Late Failure	26	61.0
Intact	3	1.2
Intact, No Vessel Breach	25	15.8

water level reaches the bottom of the recirculation lines, the operators are instructed to initiate RPV venting (to the condenser), which is classified as a late containment failure. It should be noted, that as stated in Section 4.8.2.1.11 of the submittal, the licensee is considering alternatives to the current procedure for containment flooding/RPV venting, since sensitivity analyses in the IPE submittal have shown that the current procedure has a negative impact on containment performance.

In spite of these comments, the licensee's process for the evaluation of containment failure probabilities and failure modes is consistent with the intent of Generic Letter 88-20, Appendix I. The licensee has considered the failure of the containment isolation system and containment bypass scenarios. A number of sensitivity analyses have also been performed. All of the phenomena of relevance to BWR severe accident phenomenology have been included in the submittal, as well as the principal phenomenological uncertainties. In Section 4.8.2 of the submittal, it is stated that based on recommendations in NUREG-1335 and the EPRI "Guidance Document" for using MAAP [10], the following uncertainty issues were investigated through performance of sensitivity analyses with the MAAP code:

- Core Melt Progression/In-Vessel Hydrogen Generation
- Amount of Core Debris Retained in RPV
- RPV Pressure at Vessel Failure
- Containment Pressure Load due to RPV Failure
- Direct Containment Heating
- Shell Failure by Liner Melt Through
- Debris Spread in Containment
- Ex-Vessel Debris Coolability
- Containment Failure Location
- Containment Failure Area
- Containment Flooding
- Saturated Pool Decontamination Factor

A brief description of each sensitivity case is provided in Table 4.8-3 of the submittal, and a summary of insights and conclusions from these analyses is presented in Table 4.8-15. In addition, a number of uncertainties in the Pilgrim containment performance were treated indirectly through sensitivity analyses of several CPET event probabilities. As stated in the submittal, sensitivity analyses were performed for those parameters which were judged to have large uncertainties or were expected to significantly influence the final results. The sensitivity analyses performed include the following:

- Probability of In-Vessel Cooling
- Probability of Large RPV Failure Due to Lower Head Thermal Attack
- Probability of Large RPV Failure Due to In-Vessel Steam Explosion
- Probability of Pedestal Failure Due to Overpressurization at RPV Failure
- Impact of Water in Drywell at RPV Failure
- Probability of Liner Thermal Failure at RPV Failure

- Probability of Core/Concrete Interactions
- Fraction of PDS Sequences with RPV Depressurized and With In-Vessel Injection Available
- Containment Flooding and Venting
- DCH Drywell Overpressure Failure

The principal insights from these sensitivities are provided in Section 4.8.1 of the submittal, and summarized in Section 4.8.3.1 of the submittal.

E.4 Containment Performance Improvements

Generic Letter 88-20, Supplement Numbers 1 and 3 identified specific Containment Performance Improvements (CPIs) to reduce the vulnerability of containments to severe accident challenges. For BWRs with Mark I containments, the following improvements were identified:

- Alternative water supply for drywell spray/vessel injection,

- Enhanced reactor pressure vessel depressurization system reliability,

- Implementation of Revision 4 of the BWR Owners Group EPGs, and

- Installation of a hardened vent.

At the time of issuance of the Generic Letter, the licensee had already implemented the following procedural and hardware modifications that are consistent with the recommendations of the CPI program:

- Installation of a hardened vent path.

- Modification of the existing plant systems to provide an alternate source of water injection into the vessel through the fire water cross-tie.

- Installation of a third diesel generator to enhance the reliability of AC power.

- Implementation of Revision 4 of the BWROG EPGs.

- Installation of a backup nitrogen supply system to provide long term pneumatic control capability to the Automatic Depressurization System (ADS).

In summary, the licensee's modifications are consistent with the CPI recommendations.

E.5 Vulnerabilities and Plant Improvements

Section 5.1 of the submittal provides the following criteria, used to determine if any vulnerabilities exist at the plant:

a) Are there any new or unusual means by which core damage or containment failure occur as compared to those identified in other PRA's?

b) Do the results suggest that the Pilgrim core damage frequency would not be able to meet the NRC's safety goal for core damage?

It is stated in the submittal that, based on the above criteria, no potential vulnerabilities were identified for the Pilgrim nuclear Station.

E.6 Observations

The back-end portion of the Pilgrim IPE submittal provides a substantial amount of information in regard to the subject areas identified in Generic Letter 88-20 and NUREG-1335. The PSA methodology used for the back-end analysis is basically sound, capable of identifying plant-specific vulnerabilities to release of radioactivity to the environment, and includes all the relevant phenomenological issues. The quantification of accident progression is based in large part on numerical estimates provided in NUREG-1150 supporting documentation; however, the submittal does not always contain adequate documentation as to the applicability of these estimates to the Pilgrim plant.

The important points of the technical evaluation of the Pilgrim IPE back-end analysis are summarized as follows:

- The back-end portion of this IPE submittal, for the most part, is relatively well performed and well written.
- The number of non-zero PDSs was limited to 12, and the impact of each PDS on containment performance was specifically analyzed with a CPET.
- The submittal includes all phenomena of relevance to severe accident progression for BWRs with Mark I containments.
- The submittal considers the impact of severe accident conditions on the operability of equipment.
- All CPI recommendations have been addressed, either in the submittal, or previously as part of the licensee's Safety Enhancement Program.

Several minor weaknesses (with regards to their overall impact on the IPE results) exist, and they include the following:

- The PDS definitions in the submittal are not very transparent, due to the omission of some key accident characteristics (e.g., initiating event type, or status of AC electrical power).

- In a number of instances, the submittal makes use of parameter or probability values from the Grand Gulf or Peach Bottom NUREG-1150, but does not provide a basis for their applicability to the Pilgrim plant.

- The treatment of early drywell failure due to overpressurization is inadequate.

In spite of these identified weaknesses, the licensee's process for the evaluation of containment failure probabilities and failure modes is consistent with the intent of Generic Letter 88-20, Appendix I. The dominant contributors to containment failure are consistent with the insights obtained from the NUREG-1150 analyses for the Peach Bottom plant. In summary, it is concluded that the IPE submittal provides a substantial amount of information in regard to the subject areas identified in Generic Letter 88-20 and NUREG-1335.

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NOMENCLATURE

AC	Alternating Current
ADS	Automatic Depressurization System
ATWS	Anticipated Transient Without Scram
BECo	Boston Edison Company
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CCI	Core Concrete Interactions
CDF	Core Damage Frequency
CRD	Control Rod Drive
CET	Containment Event Tree
CHR	Containment Heat Rejection
CPI	Containment Performance Improvement
DC	Direct Current
DCH	Direct Containment Heating
DG	Diesel Generator
ECCS	Emergency Core Cooling Systems
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
EVSE	Ex-Vessel Steam Explosion
FPS	Fire Protection System
GE	General Electric
GL	Generic Letter
HPME	High Pressure Melt Ejection
HRA	Human Reliability Analysis
IPE	Individual Plant Examination
ISLOCA	Interfacing Systems Loss of Coolant Accident
LOCA	Loss of Coolant Accident
LOSP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
LT-SBO	Long Term Station Blackout
MAAP	Modular Accident Analysis Program
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valve
NRC	Nuclear Regulatory Commission
PDS	Plant Damage State
PRA	Probabilistic Risk Assessment
RCS	Reactor Coolant System
RHR	Residual Heat Rejection
RPV	Reactor Pressure Vessel
SBO	Station Black-Out

NOMENCLATURE (Continued)

SORV	Stuck-Open Relief Valve
SRV	Safety Relief Valve
SSW	Station Service Water
TER	Technical Evaluation Report
TW	Loss of Decay Heat Removal
USI	Unresolved Safety Issue

1. INTRODUCTION

This Technical Evaluation Report (TER) documents the results of a review of the Pilgrim Nuclear Power Station Individual Plant Examination (IPE) Back-End submittal [1], based on the following review objectives set forth by the NRC:

- To determine if the IPE submittal essentially provides the level of detail requested in the "Submittal Guidance Document," NUREG-1335,
- To assess the strengths and weaknesses of the IPE submittal,
- To provide a preliminary list of questions based on this limited review, and
- To complete the IPE Evaluation Data Summary Sheet.

The remainder of Section 1 of this report describes the technical evaluation process employed in this review, and presents a summary of the important characteristics of the Pilgrim nuclear plant related to containment behavior and post-core-damage severe accident progression, as derived from the IPE. Section 2 summarizes the review technical findings, and briefly describes the submittal scope as it pertains to the work requirements. Each portion of Section 2 corresponds to a specific work requirement as outlined in the NRC contractor task order. A summary of the overall IPE evaluation and review conclusions are summarized in Section 3. Section 4 contains a list of cited references. Appendix A to this report contains the IPE evaluation data summary sheets.

1.1 Review Process

The technical review process for back-end analysis consists of a complete examination of Sections 1, 2, 4 through 6, and Appendices C through G of the IPE submittal. In this examination, key findings are noted; inputs, methods, and results are reviewed; and any issues or concerns pertaining to the submittal are identified. The primary intent of the review is to ascertain whether or not, and to what extent, the back-end IPE submittal satisfies the major intent of Generic Letter (GL) 88-20 [3] and achieves the four IPE sub-objectives. A draft TER based on the back-end portion of the submittal was submitted to the NRC in December 1994. A list of questions and requests for additional information was developed to help resolve issues and concerns noted in the examination process, and was forwarded to the licensee. The final TER is based on the information contained in the IPE submittal [1], and the licensee responses to the NRC Requests for Additional Information (RAIs) [10].

1.2 Containment Analysis

The Pilgrim plant is a General Electric Company BWR/3 plant with a Mark I containment located at Plymouth, Massachusetts. The design features of the Mark I containment are described in Section 4.1.1 of the submittal. The drywell, a steel pressure vessel enclosed in

reinforced basaltic concrete, is shown in Figures 4.1-2 and 4.1-3. As stated in the submittal, the drywell has a removable head which is held in place by bolts, and is sealed with a double gasket. The drywell internal design pressure is 56 psig (at a temperature of 281°F).

The suppression chamber is a torus-shaped steel pressure vessel, and is supported by the concrete foundation slab of the reactor building. Eight vent pipes connect the drywell to the suppression chamber vent header and its downcomer pipes, which discharge well below the water level in the torus.

The various containment systems considered in the Containment System Event Trees are described in Section 4.1.2 of the submittal.

2. CONTRACTOR REVIEW FINDINGS

The present review compared the Pilgrim IPE submittal to the requirements of Generic Letter (GL) 88-20, according to guidance provided in NUREG-1335. The findings of the present review are reported in this section. The review findings reported in Section 2.1 follow the structure of Task Order Subtask 1.

2.1 Review and Identification of IPE Insights

2.1.1 Completeness and Methodology

The IPE submittal contains a substantial amount of information in accordance with the recommendations of GL 88-20 and NUREG-1335.

The methodology employed in the Pilgrim submittal for the back-end evaluation is clearly described, and the IPE is logical, traceable, and consistent with GL 88-20. The definition of Plant Damage States (PDSs) involved using the core damage sequences as input to Containment Safeguards Event Trees (CSETs). The CSET nodes represent those critical safety functions which describe the post core damage status of systems important to accident progression, containment response, and radiological release. Probabilistic quantification of severe accident progression involved development of a relatively small Containment Phenomena Event Tree (CPET), and use of small, supporting fault trees to tailor each CPET question to the specific PDS being evaluated. The results of the CPET analyses lead to an extensive number of end-states, which were in turn binned into a manageable number of release categories, based on similarities in accident progression and source term characteristics.

2.1.2 As-Built/As-Operated Status

A description of the plant walkdown process undertaken at Pilgrim is provided on pages 2.1-6 and 2.1-7 of the submittal. Two stages of walkdowns are discussed. The first set of walkdowns was performed by a group of consultants as part of the limited scope Individual Plant Evaluation Methodology (IPEM) study, performed for Pilgrim prior to issuance of the generic letter. According to the submittal, the second set of walkdowns was performed by two members of the PRA team, both of which have substantial operations experience, and included the inside of the primary containment, in order to check physical parameters required for the MAAP computer code modeling.

2.1.3 Licensee Participation and Peer Review of IPE

The IPE was a cooperative utility-contractor effort, with most of the work being performed by BECo staff. Tenera, Fauske and Associates, and Gabor, Kenton, and Associates were the contractors. Five BECo senior engineers were involved in the IPE. A list of the BECo staff which participated in the performance of the Pilgrim IPE is presented in Table 2.1-3 of the submittal. It appears that BECo staff were involved in many, if not all, aspects of the study.

However, since no personnel breakdown by task is provided, it is impossible to ascertain the actual level of utility involvement. It is stated in Section 2.1.2.1 of the submittal, that BECo staff were involved in all areas of the IPE (both level 1 and level 2), and that a complete transfer of technology was accomplished. As stated in Section 2.1.2.2 of the submittal, the Pilgrim IPE underwent four levels of review. The first level of review involved a review of the IPE assumptions and results, principally performed by the Boston Edison Company (BECo) PRA/IPE staff, with limited support from consultants. The second level of review involved a review of other similar industry studies, principally insights from the limited scope IPEM, and plant-specific information from the Peach Bottom/NUREG-1150 study [2]. An internal peer review provided the third level of review for the Pilgrim IPE. As stated in the submittal, "[t]his independent, in-house review was conducted to ensure the accuracy of the documentation contained in the report, and to validate the IPE process and results." The final level of review was performed by an external peer review team, which was comprised of "industry experts who reviewed the IPE for completeness and correctness of the methods used. They also contrasted the Pilgrim results with other PRA's they were familiar with to assure consistency."

Specifically in regards to the back-end analysis, it is stated in the submittal that an independent review was performed by the primary back-end contractor. However, while Section 2.1.2.2 of the submittal lists a number of important comments/insights that were obtained from the review process, however, these comments do not pertain to the back-end portion of the submittal.

2.2 Containment Analysis

This section provides a review of PDS binning, CET analyses, release category definitions, severe accident analyses, and the containment structural analyses in the submittal.

2.2.1 Front End/Back End Dependencies

The front-end analyses in the IPE submittal report a Core Damage Frequency (CDF) of 5.85 x 10^{-5} per reactor year. The dominant contributors to core damage are high pressure core damage sequences (initiated by loss of offsite power or by transients such as loss of feedwater) with loss of coolant makeup (79.3%), followed by Anticipated Transient Without Scram (ATWS) sequences (7%), low pressure core damage sequences with loss of coolant makeup (6.2%), and LOCAs (3.2%).

After the IPE was submitted to the NRC, the licensee revised the models for CDF, including a number of enhancements. Specific examples include: elimination of the dependency of HPCI on room cooling, reduction in the number of Safety Relief Valves (SRVs) required for the success of the ADS, modification of DC power success criteria, development of recovery actions for SBO, common cause breaker failures, etc. In addition, revisions of the initial estimates for Loss of Offsite Power (LOOP) were also made in the IPE submittal. The net impact of these changes is the reduction of the CDF from 5.85 x 10^{-5} per reactor year to 2.84 x 10^{-5}. In addition, the relative contributions of the various PDSs to the CDF, were also found to be different than in the original submittal. However, the containment analyses were not revised.

The interface between the front-end (level 1) analysis and the back-end (level 2) analysis is accomplished through the propagation of front-end (core damage) sequences through the Containment System Event Trees (CSETs), as described in Section 4.3 of the submittal. As stated in the submittal, the CSETs are used to define the status of systems which are important for analyzing containment response to accident challenges. Since the actual cutsets from the level 1 analysis are input to the CSETs, system dependencies are accounted for between the level 1 systems and the containment systems. The core damage/CSET sequences are grouped together into Plant Damage States (PDSs), based on functional characteristics important to accident progression, containment failure and source term definition. As in the submittal (page 4.3-7), these characteristics are listed below by functional category.

- CONTAINMENT STATUS PRIOR TO CORE DAMAGE

  Containment Bypassed
  Containment Failed/Not Isolated

- INITIATING EVENT TYPE

  ATWS
  Non-ATWS

- CONTAINMENT MITIGATION FEATURES

  Containment Heat Removal
  Drywell Spray
  Containment Vent
  Vapor Suppression

- REACTOR PRESSURE VESSEL STATUS

  In-Vessel Injection Recovered During Core Damage
  Injection Available Following Vessel Failure
  RPV Pressure (High or Low)

In the submittal, Figure 4.3-3 presents the Plant Damage State Grouping Diagram. This diagram defines 63 different plant damage states based on the characteristics listed above. However, only 12 of the PDSs actually contain cutsets. Each PDS which contributes greater than one percent of the total core damage frequency is described in Section 4.3.3 of the submittal.

The PDS binning process appears to be reasonable and relatively complete, and includes most indicators of interest to the back-end analysis. However, the PDS definitions are deficient in two regards. First, for initiating event type, the PDS definitions only discriminate between ATWS and non-ATWS initiators. Secondly, the PDSs do not define the availability, or potential

availability, of plant AC and DC electrical power systems. Without the specific initiating event type and status of AC and DC power systems included in the PDS definitions, it is difficult to determine what operator recovery actions are possible for preventing RPV or containment failure, or for mitigating effects of a radiological release.

2.2.2 Containment Event Tree Development

Probabilistic quantification of severe accident progression for the probabilistically significant PDSs was performed using a Containment Phenomena Event Tree (CPET). The methodology employed in the Pilgrim submittal involved development of a relatively small CPET, and use of small, supporting fault trees to tailor each CPET question to the specific PDS being evaluated. The Pilgrim CPET contains the following nine nodes:

- Debris Cooled In-Vessel
- Small Lower RPV Head Failure
- No Pedestal Failure at RPV Failure
- No Drywell Failure at RPV Failure
- Debris Cooled Ex-Vessel
- No Drywell Over-Temperature Failure
- Containment Not Challenged By Over-Pressurization
- Wetwell Vapor Space Failure or Venting
- No Pool Bypass

A phenomenological fault tree was developed to support the quantification of each CPET heading listed above. A description of each phenomenological fault tree, and its quantification, is provided in Section 4.5.2 of the submittal. As stated in the submittal, the quantification of the fault tree basic events was obtained through one or more of the following sources:

1. Pilgrim-specific MAAP calculations,
2. Peach Bottom or Grand Gulf/NUREG-1150 supporting documentation [3,4], or
3. Pilgrim-specific "hand calculations."

The first node in the CPET considers the probability of recovering vessel injection and cooling the core debris in-vessel. The values used for the conditional probability of debris cooling in-

vessel are stated to have been obtained from the Grand Gulf/NUREG-1150 supporting documentation. ATWS sequences with failure to initiate standby liquid control are assumed to be not coolable in-vessel.

Given that core damage was not arrested in-vessel, the second CPET node is used to determine the size of the initial RPV lower head failure (i.e., either a small instrument tube or control rod drive penetration failure, or a large breach). Conditional probabilities of large vessel breach given an in-vessel steam explosion (IVSE) or direct thermal attack are initially taken from Grand Gulf NUREG/CR-4551, then reduced by 60 to 90 percent. It is stated in the submittal (pages 4.5-10 and -11), that these lower values for the probability of a large breach are based on recent studies, with references provided. However, no rationale is provided to support the use and applicability of this general literature for Pilgrim. Note, that the effect of assuming a higher probability of a small RPV lower head breach results in a higher probability of ex-vessel debris cooling (due to the increased probability of high pressure melt ejection).

The third CPET node assesses whether or not the pedestal fails at the time of RPV failure. In the submittal, pedestal failure at RPV failure is considered to result from either the dynamic loading associated with an ex-vessel steam explosion (EVSE), or from quasi-static overpressurization. As stated in Section 4.5.2.3, due to the relatively small depth of water resulting from flooding of the containments, it is considered unlikely that an EVSE would be of sufficient force to fail the pedestal walls, and a probability of 0.001 is assigned. However, water depth is not the most relevant parameter for determining the loading associated with an EVSE. Of more importance are parameters such as vessel failure mode, debris mass, degree of superheat, etc. Also, while it is inferred that this value is obtained from the Peach Bottom NUREG/CR-4551 study, no basis or actual origin for this value is provided. Similarly, the specific basis and origin are not provided for the value used for the probability of pedestal failure due to static overpressure given a low pressure case with a large RPV failure (page 4.5-14 of the submittal).

In addition, as discussed on page 4.5-15 of the submittal, for calculating the probability of static overpressurization of the pedestal under high pressure conditions, Peach Bottom analysis in NUREG/CR-4551 showed peak pressures ranging from 413 to 518 psid for large RPV failures, and peak pressures of 207-403 psid for small RPV failures. Pilgrim-specific MAAP calculations for these two cases are stated to indicate peak pressures of 109 psid and 30 psid, respectively. No supporting documentation is provided for the Pilgrim-specific peak pressures, nor is an explanation provided for the substantial differences in reported peak pressures between the Peach Bottom and Pilgrim analyses.

The fourth CPET node considers drywell failure at RPV failure. As described in Section 4.5.2.4 of the submittal, the four drywell failure mechanisms considered include "alpha" mode (in-vessel steam explosion), pedestal structural failure, drywell liner melt-through, and drywell overpressurization. These failure mechanisms represent all of the early drywell failures of concern for BWR Mark I containments. Drywell liner melt-through, which has typically been shown in other studies to be the dominant early containment failure mode for Mark I

containments, is of somewhat lesser importance in the Pilgrim IPE, due to the incorporation of the results of a more recent study by Theofanous, et al. [5].

The probability of cooling the core debris in the pedestal following RPV failure _ evaluated in the fifth CPET node. Values are provided for different cases depending on water availability in the cavity, the energy level of debris dispersal, and the quantity of initial debris release. It is important to note that for a large debris mass, non-energetic release to an initially dry cavity, the probability of non-coolability is assessed to be indeterminate, and therefore assigned a value of 0.5. Since, as stated in the submittal (page 4.5-23), if all debris fills the inner sumps, initial coolability is "unlikely," assigning a probability of 0.5 may be optimistic. Also, it is not apparent whether or not consideration was given to the probability that less than the maximum (or relatively little) heat might be removed from the debris bed, due to crust formation.

The sixth CPET node involves the probability of drywell over-temperature failure. As stated in the submittal (page 4.5-24), in the absence of drywell sprays and in-vessel injection following RPV failure, drywell gas temperatures will rapidly exceed 500°F, resulting in substantial leakage through the drywell head silicone closure seals.

The final three CPET nodes involve overpressure challenge to the containment from steam or non-condensible gas generation, wetwell failure or venting, and suppression pool bypass. As stated in the submittal, the containment will be subject to gradual overpressure if containment heat removal is unavailable, or if there is no ex-vessel debris cooling. Wetwell failure or venting will be prevented given drywell or RPV venting, or in the event of a drywell head overpressure failure. Suppression pool bypass (other than drywell failure or venting) is assumed to occur by either a stuck open wetwell/drywell vacuum breaker, or by loss of suppression pool inventory below the downcomer quenchers. Based on the Pilgrim containment structural evaluation, this last failure mode (i.e., loss of pool inventory) was deemed to be of insignificant probability.

In summary, the methodology employed in the Pilgrim IPE submittal is relatively well organized and easy to comprehend. However, even though the Pilgrim CPET includes most of the relevant phenomena for BWRs with Mark I containments; nevertheless, the quantification of their impact on the Pilgrim containment is relatively weak. For the most part, the quantification is based on the NUREG-1150 study, but the applicability of many of these values to the Pilgrim plant is not adequately documented in the submittal. For instance, high pressure melt ejection/direct containment heating (HPME/DCH) is only considered indirectly, by using Peach Bottom NUREG/CR-4551 calculations for the drywell pressure rise at vessel breach (page 4.5-18 of the submittal). Also, no truncation value is reported for the quantification of the PDSs; therefore, it is not possible to ascertain whether all important sequences have been analyzed through use of CPETs.

2.2.3 Containment Failure Modes and Timing

As discussed in Section 4.4 of the submittal, instead of performing a detailed structural analysis of the Pilgrim containment overpressure capacity, a review was performed of prior structural analyses of Mark I containment components, and the following potential failure locations were considered:

Drywell

Drywell Shell
Equipment Hatch
Personnel Airlock
Mechanical Penetrations
Electrical Penetrations
Drywell Head Closure

Wetwell

Drywell to Wetwell Vent Line Bellows
Wetwell Shell
- Below Downcomers
- Above Downcomers

After comparing the mean failure pressures for all major failure locations (Table 4.4-1 of the submittal), the three failure modes identified as potentially dominant were the vent line bellows, the drywell closure head, and the drywell shell. Fragility curves for each of these failure modes were constructed, and these were combined into a composite containment fragility curve (Figure 4.4-1 of the submittal). The median containment failure pressure was calculated to be 98 psig (at design temperature), at which pressure the drywell closure head contributes 36 percent to the containment failure probability, and the wetwell vent line bellows contributes 64 percent. Note, that no reference is given for the origin of the mean failure pressures given in Table 4.4-1, nor does the submittal contain any discussion as to the applicability of these values to the Pilgrim containment.

As stated in the submittal (page 4.4-4), at temperatures greater than 500°F, significant leakage will occur past the drywell closure head seal at pressures as low as 62 psig. Figure 4.4-2 of the submittal shows the Pilgrim drywell failure pressure as a function of temperature. According to the submittal, tests performed by Sandia National Laboratories indicate that the BWR Mark I containment electrical penetration assemblies will not degrade at temperatures below 700°F [6]. As such, the drywell closure head seal is judged to be the controlling failure location for containment overtemperature failure.

From information contained in Reference [7], the submittal concluded that the dominant failure locations are the drywell to wetwell vent line bellows (with a mean failure pressure of 100 psig)

and the drywell head closure (with a mean failure pressure of 125 psig). According to the submittal, the MAAP analyses performed for the Pilgrim IPE considered both of these failure modes, as well as drywell failure due to overtemperature. Also, for specific accident conditions, the drywell shell was assumed to fail due to direct contact by core debris.

In summary, the submittal appears to identify and analyze all relevant potential containment failure modes. All applicable containment failure modes from Table 2.2 of NUREG-1335 have been considered in the CPET analysis. In addition, the issue of containment overtemperature on penetration seals has been addressed. The procedure for obtaining an overall containment fragility curve using the data from Table 4.4-1 is reasonable, and standard for many IPEs. However, no reference, or basis, for the use of the mean containment failure pressures of Table 4.4-1 has been provided.

2.2.4 Containment Isolation Failure

In the Pilgrim IPE submittal, containment isolation failure is accounted for in the definition of plant damage states. In the PDS Grouping Diagram (Figure 4.3-3 in the submittal), the first two nodes are "Containment Bypassed" and "Containment Failed Prior to Core Damage." The latter node encompasses both sequences with containment failure prior to core damage, as well as sequences with loss of containment isolation. All containment bypass sequences are assigned to PDS 63, which has a frequency of 2.4×10^{-7} per reactor year (ry). Sequences with the containment failed (or not isolated) prior to core damage are included in PDSs 44 and 51, which have frequencies of 8.8×10^{-7}/ry and 4.0×10^{-6}/ry, respectively, and their frequencies were calculated using fault trees.

All containment bypass sequences were mapped directly into Release Category 34, and contribute 0.4 percent to total core damage frequency. Sequences with containment failure prior to core damage (including isolation failure) are mapped into Release Categories 17 through 25, and contribute 8.6 percent to total core damage frequency.

2.2.5 System/Human Response

The three operator actions accounted for in the CPETs are listed below:

- OPER301 - Operators turn on drywell sprays prior to RPV failure (page 4.5-13)
- OPER501(N) - Operators (fail to) initiate drywell sprays when required (pages 4.5-19 and -20)
- OPER801 - Operators initiate drywell or RPV venting (page 4.5-26)

Probabilities are not provided in the submittal for these operator actions. Adiscussion describing the situation or factors that could affect human performance is also not included.

Throughout the submittal (e.g., pages 4.8-23 to 4.8-25, pages 5.0-19 and -20, etc.), drywell and RPV venting are stated, and shown, to have a significant impact on the timing and magnitude of fission product release. Due to the importance of these actions, more details regarding venting procedures should be provided in the submittal. It is important to specify whether or not any local operator actions are required for initiation of wetwell or drywell/RPV venting, and, include an analysis of the effect of the potentially degraded environment on human performance.

Lastly, since the availability of AC electrical power is not included as part of the PDS definitions, or discussed as part of the CPET fault tree descriptions (Section 4.5.2), it is impossible to tell where, or if, credit is taken for AC power recovery.

2.2.6 Radionuclide Release Categories and Characterization

The results of the CPET analyses lead to an extensive number of end-states, which are in turn binned for source term analyses. This process is analogous to the definition of PDSs for the level 1 to level 2 interface. Outcomes of the CPETs are classified into a manageable number of release categories, which are categorized by similarities in accident progression and source term characteristics.

As discussed in Section 4.7.1 of the submittal, the sequence characteristics which were identified as having the greatest impact on fission product release at Pilgrim are the following:

- Containment Bypass
- Debris Cooled In-Vessel
- Time of Containment Failure/Venting (Relative to Core Damage)
- Mode/Location of Containment Failure (or Venting)
- Suppression Pool Bypass
- Type of Ex-Vessel Core/Concrete Interactions (Dry, Wet or None)

Using these characteristics as headings, a Source Term Category Grouping Diagram was developed (Figure 4.7-1 of the submittal). The end points of this diagram represent the individual release categories. It is interesting to note that the Pilgrim IPE definition of release categories does not consider two characteristics typically considered in other IPEs; namely, the operation of drywell sprays, and the vessel pressure at the time of RPV failure.

A total of 34 release categories are defined by the diagram in Figure 4.7-1, of which only 20 are reported as non-zero (i.e., frequency > 10^{-10}/ry). As described in Section 4.7.3 of the submittal, MAAP calculations were performed to determine the source terms for representative sequences of 12 of the 20 non-zero release categories. Of the remaining eight release categories, the source terms for seven were determined to be similar to one of the previously calculated source terms. The final release category involved no containment failure or venting, and was assigned no source term. A summary of the releases associated with each non-zero release category is provided in Table 4.7-5 of the submittal. The source terms reported in this table do not take credit for any fission product retention in the reactor building.

Generic Letter 88-20 states that "any functional sequence that has a core damage frequency greater than or equal to 10^{-6} per reactor year and that leads to containment failure which can result in a radioactive release magnitude greater than or equal to the BWR-3 or PWR-4 release categories of WASH-1400," or "any functional sequences that contribute to a containment bypass frequency of 10^{-7} per reactor year," should be reported. From Table 4.7-5 of the submittal, three release categories are identified as having frequencies greater than 10^{-6} per reactor year and having CsI releases comparable to the BWR-3 category of WASH-1400 [8]. All of these release categories (RC-14, RC-15 and RC-16) principally involve early containment failure caused by drywell liner melt-through (with the principal difference being the level of core/concrete interaction). From Table 4.7-1 of the submittal, it is seen that there are three instances of a PDS which contributes at least 10^{-6} per reactor year to core damage frequency for at least one of these release categories. Specifically, PDS-21 contributes 2.17×10^{-6}/ry to RC-16, and PDS-22 contributes 1.20×10^{-6}/ry to RC-14 and 1.10×10^{-6}/ry to RC-15. However, since the dominant functional sequence frequencies for these PDS are not reported, it cannot be ascertained as to whether or not any functional sequences exceed the reporting criteria. In addition, the containment bypass release category (RC-34) has a frequency of 2.40×10^{-7}/ry, and consists only of PDS-63. Again, however, it is impossible to tell if any "functional sequence" in PDS-63 exceeds the reporting criteria. In response to an NRC question, the licensee confirmed that it was not possible to identify functional sequences that meet the reporting criterion, due to the complexity of the methodology used.

The 34 release categories are further binned into five groups based on the magnitude of cesium iodide release. The definition of the release groupings and the conditional probability of each group (expressed as a percentage of the CDF), are provided in Table 1.

It is important to note, that the submittal (on page 4.7-14) determined that only noble gas and CsI source terms are "necessary or desirable" to be reported for the purposes of the IPE. Source terms associated with other radionuclide species are necessary for development of a full awareness of severe accident behavior. For example, the releases associated with refractory fission products are necessary for determining the extent of ex-vessel releases in BWR Mark I containments. However, it is consistent with the NRC position on the reporting criterion.

Table 1 Radionuclide Release as a Percentage of the Internal Events CDF

Release Category Grouping	Definition	Percentage of CDF
High (H)	> 10% CsI Release	14.5
Medium (M)	> 1% CsI Release	64
Low (L)	> .1% CsI Release	2.8
Low-Low (LL)	> .01% CsI Release	1.7
Negligible	Negligible Release	17

Also, as stated above, one requirement of GL 88-20 is the identification of all functional sequences with frequency greater than 10^{-6}/ry, and releases comparable to those of the WASH-1400 BWR-3 category. However, the functional sequences with frequency that meet the reporting criteria could not be determined from the submittal.

As part of this review, a comparison was made between the releases reported in Table 4.7-5 of the submittal and Figures B.3-2a and B.3-2b in Peach Bottom NUREG/CR-4551 [3]. Figure B.3-2a gives the range of expected radionuclide releases for accidents which result in early drywell failure at high RPV pressure, which roughly corresponds to Pilgrim release categories RC-14, RC-15 and RC-16. The CsI releases reported in the Pilgrim submittal for these three release categories all compare reasonably well with the mean iodine (I) release given in Figure B.3-2a for Peach Bottom. Figure B.3-2b in Peach Bottom NUREG/CR-4551 gives the range of expected radionuclide releases for accidents resulting in containment venting, which roughly corresponds to Pilgrim (non-zero) release categories RC-2 to RC-5, and RC-17 to RC-20. Again, the CsI releases reported in the Pilgrim submittal for these release categories compare reasonably well with the mean I release given in Figure B.3-2b for Peach Bottom, with the exception that those Pilgrim release categories which include suppression pool scrubbing are roughly a factor of 10 lower than those in Peach Bottom. This difference is most likely due to the fact that Pilgrim assumes a decontamination factor of 10 for saturated pool conditions.

2.3 Quantitative Assessment of Accident Progression and Containment Behavior

2.3.1 Severe Accident Progression

MAAP-BWR 3.0B Revision 7.03 was the principal tool used to analyze postulated severe accidents at Pilgrim. Even though MAAP-BWR 3.0B Revision 8.00 became available at the time of the analysis, it was not utilized since it had yet to receive widespread utility use or acceptance. As stated in the submittal, a few modifications were incorporated into the code version for the analysis. These modifications are discussed in Section 4.2.1.3.1 of the submittal. Section 4.2.1.3.3 of the submittal discusses the incorporation of EOP modelling into the MAAP analyses. This was accomplished using MIPS, the MAAP input/output processor utility [9]. The MIPS input file is listed in Appendix D, which is not provided. The MAAP plant parameter file is briefly discussed in Section 4.2.1.3.4. A listing of the actual MAAP input file is included in Appendix G, which is not provided. Section 4.6.1 of the submittal includes a description of the accident progression, and MAAP results, for the three dominant types of sequences at Pilgrim (representing over 92 percent of total core damage frequency). Summary listings of major input parameters and assumptions, and of the results, for all MAAP runs are provided in submittal Tables 4.6-7 and 4.6-8, respectively. The complete input and output for all of the MAAP analyses are included in Appendix E, which is not provided.

All of the phenomena of relevance to BWR severe accident phenomenology have been included in the submittal, as well as the principal phenomenological uncertainties. In Section 4.8.2 of the submittal, it is stated that based on recommendations in NUREG-1335 and the EPRI

"Guidance Document" for using MAAP [10], the following uncertainty issues were investigated through performance of sensitivity analyses with the MAAP code:

- Core Melt Progression/In-Vessel Hydrogen Generation
- Amount of Core Debris Retained in RPV
- RPV Pressure at Vessel Failure
- Containment Pressure Load due to RPV Failure
- Direct Containment Heating
- Shell Failure by Liner Melt Through
- Debris Spread in Containment
- Ex-Vessel Debris Coolability
- Containment Failure Location
- Containment Failure Area
- Containment Flooding
- Saturated Pool Decontamination Factor

A brief description of each sensitivity case is provided in Table 4.8-3 of the submittal, and a summary of insights and conclusions from these analyses is presented in Table 4.8-15.

2.3.2 Dominant Contributors to Containment Failure

Table 2 shows a comparison of the conditional probabilities of the containment failure modes provided in the Pilgrim IPE submittal, together with the results of the IPE submittals for the Fitzpatrick, Oyster Creek and Browns Ferry plants, as well as the NUREG-1150 study for Peach Bottom. Note, that the results reported for Peach Bottom do not include internal flood events.

From review of Table 2, it is seen that the major difference between Pilgrim and the other BWRs with Mark I containments in the table, is that Pilgrim has a significantly lower conditional probability of early containment failure, and accordingly, a much higher probability of late containment failure (with the exception of Oyster Creek). It appears that Pilgrim has understated the contribution from early drywell failure due to overpressurization (see Section 2.1.3.3 of this review). Also, for BWRs with Mark I containments, the dominant cause of early containment failure is drywell liner melt through. The submittal (p. 4.7-10) states that the principal reason for the difference in early containment failure is due to the Pilgrim assumption that for sequences with high vessel pressures at RPV failure, the probability of a liner melt through in a dry cavity is only 0.1, if in-vessel injection is available following RPV failure. Also, as stated previously in Section 2.1.2.2 of this review, the contribution of drywell liner melt through reported in the submittal, is somewhat lower due to the incorporation of the results from a more recent study. However, since the submittal does not provide the fraction of core damage frequency that is associated with a dry cavity at the time of RPV failure, as well as other event probabilities associated with the probability of liner melt through, it is not possible to determine the exact reasons why Pilgrim exhibits a much lower probability of early containment failure.

Table 2 Containment Failure as a Percentage of Internal Events CDF: Comparison to Other BWR Mark I IPEs and Peach Bottom NUREG-1150 Results

Containment Failure	Fitzpatrick IPE	Oyster Creek IPE	Browns Ferry IPE	Peach Bottom NUREG-1150	Pilgrim IPE
CDF (per year)	1.9×10^{-6}	3.2×10^{-6}	4.8×10^{-5}	4.3×10^{-6}	5.8×10^{-5}
Early Failure	60.4	16.4	55.7	46	21.6
Bypass	NA	7.3	NA	NA	0.4
Late Failure	26.0	26.4	16.0	26	61.0
Intact	2.5	0.0	18.4	3	1.2
No Vessel Breach	11.1	50.4	9.9	25	15.8

NA - Not Available

The relatively high probability of late containment failure reported in the submittal may be partially the result of the Pilgrim containment flooding strategy, which directs the operators to flood the containment if RPV water level cannot be maintained above the top of active fuel for non-ATWS sequences, or above two-thirds core height for ATWS sequences. Once containment water level reaches the bottom of the recirculation lines, the operators are instructed to initiate RPV venting (to the condenser), which is classified as a late containment failure. It should be noted, that as stated in Section 4.8.2.1.11 of the submittal, the licensee is considering alternatives to the current procedure for containment flooding/RPV venting, since sensitivity analyses have shown that the current procedure has a negative impact on containment performance.

2.3.3 Characterization of Containment Performance

The MAAP code was used to calculate some accident progression parameters (e.g., peak pressures from quasi-static overpressurization at RPV failure, in order to assess the probability of pedestal failure), but for many other important phenomena, values were obtained from the Peach Bottom NUREG/CR-4551 study (e.g., probability of drywell overpressure failure at vessel breach). More importantly, it does not appear that the submittal correctly characterized containment performance for early drywell failure due to overpressurization. As described in Section 4.5.2.4, ranges of values for the pressure rise at vessel breach were obtained from Peach Bottom, then combined with the possible range of containment pressures prior to vessel breach, and finally compared to the mean drywell failure pressure obtained for Pilgrim (125 psig). Since the resulting pressure peaks were all less than the mean drywell failure pressure, very small values were assigned to the probability of drywell overpressure failure (i.e., 0.001 for high pressure sequences, and non-zero probability for low pressure sequences). This treatment is approximate, and understates the potential for early drywell failures due to overpressurization, since it does not account for the uncertainty in the containment fragility and in the containment loading associated with individual severe accident phenomena. Even at peak pressures well below the mean failure pressure, there may be a small probability of drywell failure. Table 2.5-10 of Peach Bottom NUREG/CR-4551 shows early drywell overpressure conditional failure probabilities of between 2 and 19 percent for different PDSs (as opposed to 0.1 percent for Pilgrim). It is indicative of the fact that the Pilgrim treatment of drywell overpressure failure may be optimistic. In addition, in all cases where the containment is challenged by overpressurization, there is a finite probability of failure of both the wetwell and the drywell, as shown in Figure 4.4-1 of the submittal.

Figures 4.7-5 and 4.7-6 of the submittal show the Pilgrim total core damage frequency broken down by containment release time and failure mode, respectively. From comparison of these tables, it appears that sequences with RPV venting are classified as late releases. It is not evident from the brief description of the containment flooding/RPV venting procedures provided in the submittal (pages 4.8-23 to 4.8-25), why RPV venting should be classified as a late release.

A number of uncertainties in the Pilgrim containment performance were treated indirectly through sensitivity analyses of several CPET event probabilities. As stated in the submittal,

sensitivity analyses were performed for those parameters which were judged to have large uncertainties or were expected to significantly influence the final results. The sensitivity analyses performed include the following:

- Probability of In-Vessel Cooling
- Probability of Large RPV Failure Due to Lower Head Thermal Attack
- Probability of Large RPV Failure Due to In-Vessel Steam Explosion
- Probability of Pedestal Failure Due to Overpressurization at RPV Failure
- Impact of Water in Drywell at RPV Failure
- Probability of Liner Thermal Failure at RPV Failure
- Probability of Core/Concrete Interactions
- Fraction of PDS Sequences with RPV Depressurized and With In-Vessel Injection Available
- Containment Flooding and Venting
- DCH Drywell Overpressure Failure

The principal insights from these sensitivities are provided in Section 4.8.1 of the submittal, and summarized in Section 4.8.3.1. A brief listing of these insights is provided below:

- The probability of early containment failure is highly sensitive to assumptions regarding liner melt through.

- The probability of liner melt through is highly sensitive to the presence of water on the drywell floor at the time of RPV failure.

- The probability of core damage arrest in-vessel is highly sensitive to RPV depressurization (since, in many cases, depressurization allows in-vessel injection using low pressure systems).

- The probability of containment failure due to drywell/RPV venting is very sensitive to the EOP-directed containment flooding strategy.

- In-vessel core damage arrest is only marginally sensitive to assumptions regarding in-vessel debris coolability.

- The major level 2 results are not strongly sensitive to direct containment heating phenomena, pedestal over-pressurization at RPV failure, ex-vessel debris coolability and core/concrete attack, or initial RPV failure size.

2.3.4 Impact on Equipment Behavior

Section 4.1.2 of the submittal provides a discussion of the various containment systems, including identification of some limitations on equipment operability due to the adverse environment associated with severe accidents. The specific limitations identified include:

- Safety relief valves will be unable to operate to depressurize the reactor if containment pressure exceeds 60 psig.

- For sequences involving containment failure, the only high pressure injection system credited is the feedwater system, and the only low pressure injection systems credited are the condensate system and the fire water cross-Tie (due to uncertainties associated with equipment survivability in the reactor building following containment breach).

- Following containment failure, the only source of water credited for the drywell sprays is the fire water cross-tie.

Note, however, that no discussion is provided regarding the adverse effects of a severe accident environment on the flow of suppression pool water to heat exchangers for suppression pool cooling or to the drywell sprays (prior to containment failure).

2.4 Reducing the Probability of Core Damage or Fission Product Release

2.4.1 Definition of Vulnerability

Section 5.1 of the submittal provides the following criteria, used to determine if any vulnerabilities exist at the plant:

a) Are there any new or unusual means by which core damage or containment failure occur as compared to those identified in other PRA's?

b) Do the results suggest that the Pilgrim core damage frequency would not be able to meet the NRC's safety goal for core damage?

It is stated in the submittal that, based on the above criteria, no potential vulnerabilities were identified for the Pilgrim Station.

2.4.2 Plant Modifications

As stated in Section 6 of the submittal, prior to issuance of the generic letter, the licensee had already made a number of plant modifications as part of its Safety Enhancement Program (SEP). Of these modifications, those with the most impact on the IPE back-end analysis include:

- Installation of a hardened vent path

- Modification of existing plant systems to provide an alternate source of water injection into the vessel (or to the drywell sprays) through the fire water cross-tie (which is protected from the harsh environment of a severe accident)

- Implementation of Revision 4 of the Emergency Procedure Guidelines

In addition, as described in Section 5.3 of the submittal, a number of important insights were obtained from the back-end containment analysis, of which the three most important are briefly described below.

<u>Containment Flooding.</u> As previously described in Section 2.1.3.2 of this review, the Pilgrim EOP's require the operators to initiate containment flooding, often resulting in RPV or drywell venting, whenever adequate RPV water level cannot be maintained. Sensitivity analyses performed with MAAP "indicate that the containment flooding strategy generally results in containment radionuclide releases which are larger and earlier than for sequences where flooding is not performed." As such, the licensee is considering alternatives to the current containment flooding/venting strategy, including limited drywell flooding, as described below.

<u>Drywell Floor Flooding.</u> Since drywell liner melt through is the dominant contributor to early drywell failure, and it has been shown that the probability of liner melt through is much lower if a pool of water exists on the drywell floor at the time of RPV failure, a strategy of limited flooding of the drywell floor prior to RPV failure could greatly reduce, or essentially eliminate, the probability of drywell liner melt through. This was demonstrated in the submittal through performance of sensitivity analysis. As such, the licensee is investigating modifications to the EOP's to incorporate limited drywell flooding using the drywell sprays.

<u>RPV Depressurization.</u> Almost 62 percent of the Pilgrim core damage frequency results from high pressure sequences. Sensitivity analyses described in Section 4.8.1 of the submittal indicated that a significant increase in the probability of arresting core damage in-vessel can be obtained if RPV depressurization occurs early enough. Since the dominant cause of failing to depressurize the RPV is operator error, the licensee is considering further evaluation of the operator error rates assumed for the analysis, and possibly modifications to procedures or operator training.

2.5 Responses to the Recommendations of the CPI Program

Generic Letter 88-20 supplement number 1 [11] and number 3 [12] identified specific Containment Performance Improvements (CPIs) to reduce the vulnerability of containments to severe accident challenges. For BWRs with Mark I containments, the following improvements were identified:

- Hardened vent,

- Alternative water supply for drywell spray/vessel injection,

- Enhanced reactor pressure vessel depressurization system reliability, and

- Implementation of Revision 4 of the BWR Owners Group EPGs.

At the time of issuance of the Generic Letter, the licensee had already implemented the following procedural and hardware modifications that are consistent with the recommendations of the CPI program:

- Installation of a hardened vent path.

- Modification of the existing plant systems to provide an alternate source of water injection into the vessel through the fire water cross-tie.

- Installation of a third diesel generator to enhance the reliability of AC power.

- Implementation of Revision 4 of the BWROG EPGs.

- Installation of a backup nitrogen supply system to provide long term pneumatic control capability to the Automatic Depressurization System (ADS).

In summary, all CPI recommendations have been addressed, either in the submittal, or previously as part of the licensee's Safety Enhancement Program.

3. OVERALL EVALUATION AND CONCLUSIONS

The back-end portion of the Pilgrim IPE submittal provides a substantial amount of information in regard to the subject areas identified in Generic Letter 88-20 and NUREG-1335. The methodology used for the back-end analysis is basically sound, capable of identifying plant-specific vulnerabilities to release of radioactive material to environment, and includes the relevant phenomenological issues. The quantification of accident progression is based in large part on numerical estimates provided in NUREG-1150 supporting documentation; however, the submittal does not always contain adequate documentation as to the applicability of these estimates to the Pilgrim plant.

The important points of the technical evaluation of the Pilgrim IPE back-end analysis are summarized as follows:

- The back-end portion of this IPE submittal, for the most part, is relatively well performed and well written.
- The number of non-zero PDSs was limited to 12, and the impact of each PDS on containment performance was specifically analyzed with a CPET..
- The submittal includes all relevant phenomena of interest to severe accident phenomenology for BWRs with Mark I containments.
- The submittal considers the impact of severe accident conditions on the operability of equipment.
- All CPI recommendations have been addressed, either in the submittal, or previously as part of the licensee's Safety Enhancement Program.

Several weaknesses (with regards to their overall impact on the IPE results) exist, and they include the following:

- The PDS definitions used in the submittal are not very transparent, due to the omission of some key accident characteristics (e.g., initiating event type, or status of AC power). This makes it impossible to determine whether AC power is available for all sequences in a PDS, and whether AC power recovery has been credited for sequences involving loss of AC power.
- In a number of instances, the submittal makes use of parameters or probability values from the Grand Gulf or Peach Bottom NUREG-1150 studies, but does not provide a basis for their applicability to the Pilgrim plant.
- The treatment of the probability of early drywell failure due to overpressurization in the submittal is weak and poorly documented.

- Generic Letter 88-20 states that "any functional sequence that has a core damage frequency greater than or equal to 10^{-6} per reactor year and that leads to containment failure which can result in a radioactive release magnitude greater than or equal to the BWR-3 or PWR-4 release categories of WASH-1400," or "any functional sequences that contribute to a containment bypass frequency of 10^{-7} per reactor year," should be reported in the IPE submittal. The licensee did not identify sequences that meet with the reporting criteria.

In spite of these identified weaknesses, the licensee's process for the evaluation of containment failure probabilities and failure modes is consistent with the intent of Generic Letter 88-20, Appendix I. The dominant contributors to containment failure are consistent with the insights obtained from the NUREG-1150 analyses for the Peach Bottom plant. The licensee has considered the failure of the containment isolation system and containment bypass scenarios. A number of sensitivity analyses have also been performed. In summary, it is concluded that the IPE submittal provides a substantial amount of information in regard to the subject areas identified in Generic Letter 88-20 and NUREG-1335.

4. REFERENCES

1. "Pilgrim Nuclear Power Station Individual Plant Examination for Internal Events per GL-88-20," Boston Edison Company, September 1992.

2. USNRC, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, Vols. 1 and 2, December 1990.

3. "Evaluation of Severe Accident Risks: Peach Bottom, Unit 2," NUREG/CR-4551, Vol. 4, Rev. 1, December 1990.

4. "Evaluation of Severe Accident Risks: Grand Gulf, Unit 1," NUREG/CR-4551, Vol. 6, Rev. 1, December 1990.

5. T. G. Theofanous, et al., "The Probability of Liner Failure in a Mark-I Containment," NUREG/CR-5423, August 1991.

6. FAI/91-76, "Thermal Attack of Containment Penetrations," Fauske & Associates, Inc., May 1991.

7. FAI/91-84, Rev. 2, "Containment Overpressurization," Fauske & Associates, Inc., January 1992.

8. USNRC, "Reactor Safety Study -- An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400 (NUREG-75/014), October 1975.

9. Gabor, Kenton & Associates, Inc., "MIPS: An Improved Input and Output Processor for the MAAP Code," Version 1.80, May 1992.

10. Gabor, Kenton & Associates, Inc., "Recommended Sensitivity Analyses for an Individual Plant Examination Using MAAP 3.0B," EPRI TR-100167, 1991.

11. NRC Letter to All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities, "Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR §50.54(f)," Generic Letter 88-20, Supplement No. 1, dated August 29, 1989.

12. NRC Letter to All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities, "Completion of Containment Performance Improvement Program and Forwarding of Insights for Use in the Individual Plant Examination for Severe Accident Vulnerabilities - Generic Letter No. 88-20 Supplement No. 3 - 10 CFR §50.54(f)," Generic Letter 88-20, Supplement No. 3, dated July 6, 1990.

13. Response to Request for Additional Information Regarding the Pilgrim Individual Plant Examination (IPE) Submittal (TAC No. M74451), Enclosure to BECo Ltr. #95-127 from E. T. Boulette, Boston Edison Company, to U.S. Nuclear Regulatory Commission Dated December 28, 1995.

APPENDIX

IPE EVALUATION AND DATA SUMMARY SHEET

BWR Back-End Facts

Plant Name

Pilgrim Nuclear Power Station

Containment Type

Mark I

Unique Containment Features

None found

Unique Vessel Features

None found

Number of Plant Damage States

63 (12 non-zero)

Containment Failure Pressure

98 psig (median)

Additional Radionuclide Transport and Retention Structures

No credit taken for retention in the Reactor Building or other structures

Conditional Probability That the Containment Is Not Isolated

0.090 (includes containment failure prior to core damage)

Important Insights, Including Unique Safety Features

MAAP sensitivity analyses indicate that the current strategy for containment flooding (and RPV venting) results in radionuclide releases which are larger and earlier than if flooding is not performed.

Implemented Plant Improvements

Consideration is being given to alternative containment flooding strategies (e.g., limited drywell flooding), and to increasing operator reliability of RPV depressurization.

Note: Plant modifications in response to all CPI recommendations have already been made as part of the BECo Safety Enhancement Program (SEP).

C-Matrix

See Table A.1 (attached)

Table A.1 Pilgrim C-Matrix

		Plant Damage State												
		1	2	9	10	15	21	22	23	25	44	51	63	Total
Release Category	1	.056	.058											.012
	2					.518								.008
	3	.056	.037			.467								.019
	4					.001								<.001
	5													<.001
	7			.518	.521			.475	.477	.468				.294
	8		.021	.469	.467		.396	.429	.432	.425				.290
	14	.075	.072	.007	.006	.007		.050	.048	.056				.048
	15	.074	.072	.007	.005	.007		.046	.043	.051				.045
	16						.604							.027
	17											.127		.009
	18											.127		.009
	19													<.001
	20													<.001
	22											.373		.026
	23											.373		.026
	24										.747			.011
	25										.253			.004
	26	.739	.737											.158
	34												1.000	.004
Totals		1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000