APPENDIX A

PILGRIM NUCLEAR POWER STATION INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(FRONT-END)

SEA 93-553-13-A:4
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Pilgrim

Technical Evaluation Report on the Individual Plant Examination Front End Analysis

NRC-04-91-066, Task 13

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E. EXECUTIVE SUMMARY

This report summarizes the results of our review of the front-end portion of the Individual Plant Examination (IPE) for Pilgrim. This review is based on information contained in the IPE submittal [IPE Submittal] along with the licensee's responses [RAI Responses] to a request for additional information (RAI).

In responding to the RAI, the licensee states that the original IPE analysis, as described in the IPE submittal, has been updated. The licensee response to the RAI describes the results of the 1995 IPE model, including updated accident sequences and dominant core damage contributors. To the extent possible, the IPE results and findings reported in this review are based on the 1995 IPE model as reported in the RAI responses.

E.1 Plant Characterization

The Pilgrim Nuclear Power Station consists of a single boiling water reactor (BWR) 3 Mark 1 unit. Design features at Pilgrim that impact the core damage frequency (CDF) relative to other BWRs are as follows:

- Fourteen hour battery capacity. With credit for load shedding, the batteries can provide necessary power during station blackout for approximately 14 hours. The 14 hour battery lifetime is longer than battery lifetimes at many other plants. This design feature tends to lower the CDF.

- Station blackout diesel generator. A station blackout diesel generator has been installed at Pilgrim. This design feature tends to lower the CDF.

- Special 23 Kv offsite power line for plant shutdown functions. Pilgrim has a special 23 Kv offsite power connection that can be used to power emergency buses in the event offsite power from the two 345 Kv sources is lost. The 23 Kv power line is routed through a separate switchyard into the station shutdown transformer. Plant experience has shown that the 23 Kv line is more resistant to weather-related effects than the 345 Kv sources. This design feature tends to lower the CDF.

- Ability to perform vessel injection with fire water system. Alternate vessel injection can be accomplished with the fire water system. Because one of the fire water pumps is diesel-driven, this method of injection can be used during station blackout. This design feature tends to lower the CDF.

- Limited vessel pressure relief capability. Pilgrim has a more limited vessel pressure relief capability than other BWRs of similar design. This limited pressure relief capability is due to the relatively small number of relief valves (2 code safety valves and 4 safety relief valves) and the relatively small capacity of each individual valve. This design feature tends to increase the CDF.

- Hardened torus vent. The availability of a hardened torus vent provides an additional means of providing containment pressure control and decay heat removal. This design feature tends to lower the CDF.

- Portable diesel-driven air compressor. A portable diesel-driven air compressor can be manually connected to the compressed air system. This additional source of compressed air tends to reduce the CDF.

- Diverse instrument nitrogen supplies. Diverse means are available for supping nitrogen to support important functions, for example the automatic depressurization system (ADS) valves. Sources of nitrogen include banks of bottled nitrogen and a trailer-mounted set of liquid nitrogen tanks. This design feature tends to reduce the CDF.

- Independence of diesel generators from external cooling water sources. The diesel generators (including the station blackout diesel generator) are self-cooled. This design feature lowers the CDF.

E.2 Licensee's IPE Process

The licensee developed a Level 2 probabilistic risk assessment (PRA) in response to the requests of Generic Letter 88-20. The freeze date for the original IPE analysis reported in the submittal was December 31, 1991. The freeze date for the updated (1995) IPE was not provided.

The licensee provided the overall technical management of the IPE and was involved in all aspects of the analysis. It appears that the licensee performed the majority of the front-end analysis. Tenerra, L. P. assisted the licensee in the front-end accident sequence evaluation.

Plant walkdowns were used to support the IPE analysis. Major documentation used in the IPE included: engineering drawings, system descriptions, the Updated Final Safety Analysis Report (UFSAR), Technical Specifications, and plant procedures.

There were several levels of review performed on the IPE, including an external peer review. The external review team consisted of 5 outside individuals with backgrounds in PRA, operations, reactor engineering, and thermal hydraulics analysis.

The licensee states that the IPE is a living model that is updated to reflect changes to plant configuration and performance.

E.3 Front-End Analysis

The methodology chosen for the Pilgrim IPE front-end analysis was a Level 1 PRA. The small event tree/large fault-tree technique with fault tree linking was used to quantify core damage sequences.

Core damage was defined to occur when the water level remained below 2/3 of the core height for 10 minutes. It appears that the front-end system success criteria were largely based on Modular Accident Analysis Program (MAAP) calculations. General Electric apparently generated some portions of the anticipated transient without scram (ATWS) success criteria with a "TRAGG" analysis. The Pilgrim IPE success criteria are generally consistent with success criteria typically used in other BWR IPE/PRA studies.

The IPE analyzed typical generic and transient initiating events, as well as 6 special initiating events representing support system failures. Plant data and plant-specific logic models were used to support the quantification of initiating events.

Plant-specific data were used where possible for component failure rates and test/maintenance unavailabilities. Component unavailability estimates were derived for the period between 01/01/81 and 09/30/89 with one major exception. The data used to quantify the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems came from a five year moving average data base for the period between 03/31/87 and 03/31/92. These two systems were quantified with the more recent data to reflect their improved availability since 1990. The data collection period of 01/01/81 to 09/30/89 was used to ensure the equivalent of 5 full years of plant operation. Pilgrim was shutdown for most of 1986 and all of 1987 and 1988 for reliability and safety enhancements.

The multiple Greek letter (MGL) method was used to model common cause failures. The common cause data used in the IPE are generally consistent with generic values typically used in other IPE/PRA studies.

The point estimate CDF from the updated (1995) IPE is 2.84E-05/yr[1], including internal flooding. The CDF contribution from flooding is 6.1E-08/yr. The internal initiating events that contribute most to the CDF and their percent contribution are listed below[2]:

Partial loss of offsite power (LOSP)	30%
Manual shutdown	19%
Full LOSP (345 & 23 Kv)	10%
Turbine trip and reactor trip	10%
Loss of feedwater	8%
Medium loss of coolant accident (LOCA)	6%
Loss of condenser vacuum	4%
Loss of DC Bus B	3%
Main steam isolation valve (MSIV) closure	3%

Core damage contributions by accident type are listed below:

Transient	70%
Anticipated Transient Without Scram (ATWS)	16%
LOCAs	10%
Station Blackout	3%
Interfacing systems LOCA	0.4%
Internal Flood	0.2%

The front and back-end analyses were coupled by linking core damage cut sets directly into containment system event trees (CSETs). The output of the CSETs was used to generate a set of plant damage states

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[1] As used here and in other portions of this report, the term "yr" refers to a reactor year.

[2] A complete set of initiating event CDF contributors is provided in Table 2-8 of this report.

(PDSs). The PDSs were subsequently evaluated by containment phenomenological event trees (CPETs) to generate source term frequencies. The process used to couple the front- and back-end analyses appears to be comparable with similar processes used in other PRA/IPE studies.

E.4 Generic Issues

The licensee specifically addresses decay heat removal (DHR) and its contribution to CDF. The IPE DHR analysis was based on a narrow DHR definition, namely removal of decay heat from containment. This definition does not address core cooling aspects of DHR. Using qualitative discussions, the licensee concluded that the containment DHR reliability is high due to the availability of multiple containment systems. Even if all these containment DHR systems were to fail, a significant amount of time is available for repair and recovery efforts. Specifically, 34 hours would be available before containment design limits are exceeded, and 47 hours would be available before the ultimate containment capacity is reached. No DHR-related vulnerabilities were identified by the licensee.

No generic safety issues (GSIs) or unresolved safety issues (USIs) other than A-45 are addressed by the IPE.

E.5 Vulnerabilities and Plant Improvements

The licensee used the following criteria to search for vulnerabilities:

- Are there any new or unusual means by which core damage or containment failures occur as compared to those identified in other PRAs?
- Do the results suggest that the Pilgrim core damage frequency would not be able to meet the NRC's safety goal for core damage (1E-04/yr)?

Based on the above criteria, the licensee concluded that there are no vulnerabilities at Pilgrim.

It appears that the following two plant improvements were identified as a result of the IPE:

- Modify loss of DC procedures to allow operator judgment for load shedding of AC buses associated with failed DC supplies (completed)

- Modify procedures to allow operators to use fire water for drywell sprays

The DC procedure enhancement has been completed. The status of the drywell spray enhancement is not known.

E.6 Observations

The licensee appears to have analyzed the design and operations of Pilgrim to discover instances of particular vulnerability to core damage. It also appears that the licensee has: developed an overall appreciation of severe accident behavior; gained an understanding of the most likely severe accidents at Pilgrim; gained a quantitative understanding of the overall frequency of core damage; and implemented changes to the plant to help prevent and mitigate severe accidents.

Strengths of the IPE are as follows: The IPE goes beyond the bounds of some other BWR IPE/PRA studies by considering and modeling common cause failures between the HPCI and RCIC systems.

No major weaknesses of the IPE were identified.

One weakness of the <u>submittal</u> was identified, namely that the licensee's DHR analysis was limited to removal of decay heat from containment. This narrow definition of DHR does not address core cooling aspects of DHR. In order to resolve USI A-45, licensees were requested to examine DHR for its capability during both core cooling and containment heat removal phases and for all accidents except large LOCAs, ATWS events, and ISLOCAs. While the licensee's narrow definition of DHR is judged to be a weakness of the <u>submittal</u>, both core and containment cooling has been accounted for in the overall IPE analysis process. In our judgment, the IPE models are capable of identifying DHR-related vulnerabilities.

Significant level-one IPE findings are as follows:

- The Pilgrim plant is located in a region of the country that is prone to more frequent occurrences of severe weather than many other nuclear plant sites. Consequently, LOSP frequencies and non-recovery probabilities are higher at Pilgrim compared to average industry data. Even so, station blackout contributes only about 3% of the total CDF at Pilgrim. The relatively low CDF contribution of station blackout at Pilgrim is due to: (1) a 14 hour battery capacity (with credit for load shedding), (2) the availability of a station blackout diesel generator, (3) the availability of a separate 23 Kv offsite power source for plant shutdown functions, (4) the availability of an AC-independent source of vessel injection (fire water), and (5) credit for recovery of failed diesel generators.

- ATWS sequences contribute 16% to the total CDF. About 55% of the ATWS contribution is due to operator failure to initiate standby liquid control (SLC) injection. Another 20% of the ATWS contribution is related to inadequate pressure relief caused by failure of sufficient safety valves/safety relief valves to open.

- Common cause failures of safety relief valves (SRVs) are important contributors in sequences where high pressure injection is unavailable and depressurization fails. As previously noted in the discussion on plant features, Pilgrim has a more limited vessel pressure relief capability than other BWRs of similar design.

1. INTRODUCTION

1.1 Review Process

This report summarizes the results of our review of the front-end portion of the IPE for Pilgrim. This review is based on information contained in the IPE submittal [IPE Submittal] along with the licensee's responses [RAI Responses] to a request for additional information (RAI).

In responding to the RAI, the licensee states that the original IPE analysis, as described in the IPE submittal, has been updated. The licensee response to the RAI describes the results of the 1995 IPE model, including updated accident sequences and dominant core damage contributors. To the extent possible, the IPE results and findings reported in this review are based on the 1995 IPE model as reported in the RAI responses.

1.2 Plant Characterization

The Pilgrim Nuclear Power Station consists of a single BWR 3 Mark 1 unit located on Cape Cod Bay in Massachusetts. Bechtel was the architect engineer and constructor. Pilgrim began commercial operation in December 1972. The plant was shutdown for most of 1986 and all of 1987 and 1988 for reliability and safety enhancements. The Monticello plant is similar to Pilgrim. [p. A-2 of submittal, pp. 1.1-2, 2.2-1 of UFSAR]

Design features at Pilgrim that impact the core damage frequency (CDF) relative to other BWRs are as follows: [pp. 1.1-4 to 1.1-6, 6.0-6, 6.0-7 of submittal]

- Fourteen hour battery capacity. With credit for load shedding, the batteries can provide necessary power during station blackout for approximately 14 hours. The 14 hour battery lifetime is longer than battery lifetimes at many other plants. This design feature tends to lower the CDF. [pp. 2.2-6, B.11-1, C.2-16, C.2-17, p. 3 of Table 2.4-1]

- Station blackout diesel generator. A station blackout diesel generator has been installed at Pilgrim. This design feature tends to lower the CDF. [p. 2.3-29 of submittal]

- Special 23 Kv offsite power line for plant shutdown functions. Pilgrim has a special 23 Kv offsite power connection that can be used to power emergency buses in the event offsite power from the two 345 Kv sources is lost. The 23 Kv power line is routed through a separate switchyard into the station shutdown transformer. Plant experience has shown that the 23 Kv line is more resistant to weather-related effects than the 345 Kv sources. This design feature tends to lower the CDF. [pp. 2.2-5, 2.3-27, 2.3-28, B.9-1, B.9-2, B.10-1, B.10-2, NRC memo from M. Rubin to A. Thadani]

- Ability to perform vessel injection with fire water system. Alternate vessel injection can be accomplished with the fire water system. Because one of the fire water pumps is diesel-driven, this method of injection can be used during station blackout. This design feature tends to lower the CDF. [pp. 2.2-6 of submittal]

- Limited vessel pressure relief capability. Pilgrim has a more limited vessel pressure relief capability than other BWRs of similar design. This limited pressure relief capability is due to the relatively small number of relief valves (2 code safety valves and 4 safety relief valves) and the relatively small capacity of each individual valve. This design feature tends to increase the CDF.

- Hardened torus vent. The availability of a hardened torus vent provides an additional means of providing containment pressure control and decay heat removal. This design feature tends to lower the CDF. [pp.2.2-6, 3.6-12, B.6-8, B.6-9 of submittal]

- Portable diesel-driven air compressor. A portable diesel-driven air compressor can be manually connected to the compressed air system. This additional source of compressed air tends to reduce the CDF. [pp. 2.2-7, 3.3-5, A1-5 of submittal]

- Diverse instrument nitrogen supplies. Diverse means are available for supping nitrogen to support important functions, for example the automatic depressurization system (ADS) valves. Sources of nitrogen include banks of bottled nitrogen and a trailer-mounted set of liquid nitrogen tanks. This design feature tends to reduce the CDF. [pp. 22.-7, B. 13-2, B.13-3 of submittal]

- Independence of diesel generators from external cooling water sources. The diesel generators (including the station blackout diesel generator) are self-cooled. This design feature lowers the CDF. [p. 9 of Table 2.4-2 of submittal]

2. TECHNICAL REVIEW

2.1 Licensee's IPE Process

We reviewed the process used by the licensee with respect to: completeness and methodology; multi-unit effects and as-built, as-operated status; and licensee participation and peer review.

2.1.1 Completeness and Methodology.

The submittal is complete with respect to the type of information requested by Generic Letter 88-20 and NUREG 1335. No omissions were noted.

The Pilgrim IPE is a level 2 PRA. The IPE used the small event tree/large fault tree methodology with fault tree linking to perform the level 1 analysis. The Cut Set and Fault Tree Analysis (CAFTA) computer code was used to quantify accident sequences. The licensee had performed an earlier Industry Degraded Core Rulemaking Program (IDCOR) Individual Plant Evaluation Methodology (IPEM) study in 1988 to support a safety enhancement program, and this earlier study was used as the starting point for the IPE. [transmittal letter, p. 2.1-1 of submittal]

Intersystem dependencies are discussed and tables of system dependencies are provided. Data for quantification of the models are provided, including common cause events and human recovery actions. The application of the technique for modeling internal flooding is described in the submittal. Results of an importance analysis of key common cause and human event CDF contributors are presented. Two types of sensitivity analysis were performed on the IPE results.

2.1.2 Multi-Unit Effects and As-Built, As-Operated Status.

The Pilgrim plant is a single unit site; therefore, multi-unit considerations do not apply to this plant.

The licensee performed walkdowns and used various sources of plant-specific information to support the analysis, for example engineering drawings, system descriptions, the UFSAR, Technical Specifications, and plant procedures. Plant records were reviewed to develop plant-specific behavioral characteristics such as component failure rates and initiating event frequencies. [pp. 2.1-5 to 2.1-9 of submittal]

The freeze date of the original IPE analysis reported in the submittal was December 31, 1991. The freeze date for the updated (1995) IPE was not provided. [pp. 2.2-2 of submittal]

The licensee states that the IPE is a living model that is updated to reflect changes to plant configuration and performance. [p. 1 of RAI Responses]

2.1.3 Licensee Participation and Peer Review.

The licensee provided the overall technical management of the IPE and was involved in all aspects of the analysis. It appears that the licensee performed the majority of the front-end analysis. Tenerra, L. P.

assisted the licensee in the accident sequence evaluation. Gabor, Kenton, and Assoc., Inc., and Fauske and Assoc, Inc. assisted with the back-end analysis. [pp. 2.1-4, 2.1-19, 2.1-20 of submittal]

Licensee personnel assigned to the IPE project included individuals with PRA expertise and licensed senior reactor operators (SROs). Other experience areas represented by the licensee IPE personnel included operator training, thermal hydraulics, operations, quality assurance, and system engineering. [pp. 2.1-18, 2.1-19, 2.1-27 to 2.1-29 of submittal]

There were several levels of review performed on the IPE. Initially, the licensee reviewed the consistency and correctness of assumptions and results, with minor contributions by consultants. This first level of review was done to ensure a complete transfer of technology to the licensee. [p. 2.1-20 of submittal]

An independent internal peer review was performed to ensure the accuracy of the documentation contained in the report and to validate the IPE process and results. This peer review team consisted of 7 individuals with backgrounds in PRA, engineering, operations, training, licensing and management. [pp. 2.1-28, 2.1-29 of submittal]

An external peer review was also performed on the IPE. The external review team consisted of 5 outside individuals with backgrounds in PRA, operations, reactor engineering, and thermal hydraulics analysis. These individuals were associated with the following organizations: Yankee Atomic Electric, Northeast Utilities, New Hampshire Yankee, Tenerra, and Gabor, Kenton, and Assoc. [pp. 2.1-4, 2.1-23, 2.1-30 of submittal]

The submittal provides examples of review comments generated by the internal and independent external review teams. Resolutions to some of these comments are also provided. [pp. 2.1-21 to 2.1-24 of submittal]

2.2 Accident Sequence Delineation and System Analysis

This section of the report documents our review of both the accident sequence delineation and the evaluation of system performance and system dependencies provided in the submittal.

2.2.1 <u>Initiating Events</u>.

Initiating events were identified from reviews of operating histories for Pilgrim and other plants, reviews of previous risk analyses, plant-specific system analyses, and reviews of generic initiating event lists (for example, EPRI NP-2230). The initiating events included in the accident sequence analysis are listed below: [p. 22 of RAI Responses, pp. 2.1-8, 2.1-9, 3.1-1 to 3.1-13, Tables 3.1-1 and 3.1-3 of submittal]

Generic Transients:

- Reactor and turbine trip
- Loss of feedwater
- Loss of feedwater (unrecoverable)
- Loss of condenser vacuum
- MSIV closure
- Inadvertent open relief valve (IORV)

Manual shutdown
Loss of 345 Kv power
Loss of all offsite power (345 Kv and 23 Kv)

Special Initiators:
Loss of DC Bus A
Loss of DC Bus B
Loss of salt service water (SSW)
Loss of reactor building cooling water (RBCCW)
Loss of turbine building cooling water (TBCCW)
Loss of instrument air

LOCAs:
Small LOCA (mitigable by RCIC)
Medium LOCA
Large LOCA
Reactor vessel rupture
Reference line break
Large LOCA outside containment (main steam line break)

Interfacing systems LOCA (ISLOCA):
Core spray interfacing systems ISLOCA
Low pressure coolant injection (LPCI) ISLOCA

Internal Flooding:
(Number of initiating events not specified)

The manual shutdown category includes all those events in which a manual scram or manual shutdown was performed for either a planned or forced outage and for which none of the other transient categories was appropriate. [p. 14 of RAI Responses]

Two separate initiating events were used to model loss of normal power, specifically loss of all offsite power (345 Kv and 23 Kv), and loss of 345 Kv. Pilgrim is connected to the power grid via the 345 Kv connection. A separate 23 Kv source is available to provide power to the station shutdown transformer. The shutdown transformer in turn can provide power to the essential 4,160 VAC buses. [pp. 2.3-27, 2.3-28, B.9-1, B.9-2, B.10-1, B.10-2]

Neither loss of an AC bus or loss of heating, ventilating, and air conditioning (HVAC) was modeled as an initiating event. The submittal does not indicate a reason for omitting these potential initiating event categories. It is noted that HVAC is used to support equipment operation in a number of plant locations.

Plant data used to support the quantification of initiating events were generally based on a collection period from 01/01/75 to 9/30/89. The data collection start date of 01/01/75 excludes Pilgrim's first year of operation to eliminate effects of the plant startup learning curve and equipment break-in. Where initiating event frequencies have been influenced by changes to plant design and operation (for example, LOSP), adjustments have been made. [p. A-2 of suhmittal]

Transient events were quantified where possible with plant data via the process described above. The main sources of plant-specific transient data were scram reports and Pilgrim Monthly Reports to the NRC. Small,

medium, and large LOCA initiating events were apparently quantified from an IDCOR BWR methodology study [IPE BWR Method]. This IDCOR study is in turn at least partially based on EPRI data [EPRI 438]. The ISLOCA and LOCA outside containment events were based on NUREG/CR-2129, BWR Owners Group guidance (no reference provided), and plant-specific considerations. [p. 7 of RAI Responses, pp. 2.1-22, 3.1-4 to 3.1-10, Tables 3.1-1, 3.1-3, A-7 of submittal]

Fault tree logic models were used to quantify the following plant-specific initiating events: loss of salt service water (SSW), loss of reactor building cooling water (RBCCW), loss of turbine building cooling water (TBCCW), and loss of instrument air. Plant-specific data were used where possible to support the quantification of these logic models. [p. 11 of RAI Responses, pp. A-8 to A-10, Table 3.1-3 of submittal]

The IPE used a frequency of 3.89/yr for manual shutdowns, based on 47 events over about 12 years of reactor operation. This value reflects a relatively high level of plant shutdowns experienced during 1975-1989, though recent plant history indicates far fewer shutdowns. [pp. 14, 15 of RAI Responses]

A frequency of 0.142/yr was used for total LOSP (345 Kv and 23 Kv) in the original IPE reported in the submittal, and apparently was also used in the 1995 IPE. If more recent plant experience through August 31, 1995 is accounted for, the total LOSP frequency becomes 0.135/yr, based on 4 LOSP events over 20.7 years of reactor operation. For partial loss of power (345 Kv), the IPE used a frequency of 0.475/yr. If plant experience through August 31, 1995 is included, the loss of 345 Kv has an initiating frequency of 0.643/yr. [p. 12 of RAI Responses]

The average frequency of total LOSP over all plants as reported in an Electric Power Research Institute (EPRI) publication [NSAC-147] is about 0.06/yr, compared to the IPE value of 0.142/yr. The higher value of total LOSP at Pilgrim can be attributed to severe weather activity experienced at Pilgrim.

The IPE frequency of 7E-04/yr for large LOCAs is a factor of 2 to 7 higher than corresponding data typically used in other BWR IPE/PRA studies. On the other hand, the Pilgrim IPE used a value of 8E-03/yr for small LOCAs, whereas some other BWR IPE/PRA studies have used values that are 5 to 10 times higher. The small LOCA frequency used in the IPE is based on EPRI guidance [EPRI 438] that suggests adding a contribution for recirculation pump seal leakage to the baseline value of 8E-03/yr, depending on the plant's vulnerability to seal leakage. The licensee states that Pilgrim's seal design is not prone to leakage, and therefore the EPRI small LOCA baseline frequency of 8E-03/yr was used. [p. 7 of RAI Responses]

The quantification of the remaining initiating events appears to be generally consistent with other PWR IPE/PRA studies.

2.2.2 Event Trees.

The following general categories of event tree models were used in the analysis: [Appendix C of submittal]

Transient and special initiators
LOSP
Station blackout and offsite power recovery
Stuck open relief valve (post-transient)

Inadvertent open relief valve
Reference leg break
Small LOCA
Medium LOCA
Large LOCA
Reactor vessel rupture
Large LOCA outside containment (main steam line break)
ISLOCA (LPCI and core spray)
ATWS (with and without isolation from main condenser)

Core damage was defined to occur when the water level remained below 2/3 of the core height for 10 minutes. It appears that the front-end system success criteria were largely based on Modular Accident Analysis Program (MAAP) calculations. General Electric apparently generated some portions of the ATWS success criteria with a "TRAGG" analysis [GE TRAGG]. The MAAP code was also used to assess containment accident progression. [pp. 38, 41 of RAI Responses, pp. 1.0-8, 2.3-1 of submittal]

The IPE assumed that vessel failure would occur with a frequency of 1E-05/yr, including both mitigable and non-mitigable breaks. Using the 1988 IDCOR Individual Plant Methodology (IPEM) for Pilgrim, the licensee assumed that mitigable medium or large LOCA breaks would occur 97% of the time. The remaining 3% of vessel breaks were assumed to be too large for successful mitigation. Therefore, the core damage frequency for non-mitigable vessel rupture events was 3E-07/yr. The vessel rupture event tree represented in Figure C.3-4 of the submittal displays the various accident sequence paths. [pp. 3.4-9, C.3-10 to C.3-12, Figure C.3-4 of submittal]

The IPE credited use of the fire water system for alternative vessel injection during transient events, small LOCAs, and reference line breaks. This action, which is proceduralized, requires installation of a spool piece to connect the fire water system to RHR system discharge piping[3]. The spool piece is located in a cabinet immediately adjacent to its installation point, and can be connected without special tools due to the use of quick snap couplings. The fire water pumps, one diesel-driven and one electric motor-driven, provide water from two 250,000 gallon onsite fire water storage tanks. Each of these pumps has a rated capacity of approximately 2,000 gpm. The licensee states that operators are trained and tested in the use of the fire water system. In addition to installation of the spool piece, use of the fire water system requires that operators open two valves. The availability of DC power is also required to allow vessel depressurization via the safety relief valves (SRVs). [pp. 23, 34, 37 of RAI Responses, pp. 2.3-15, 2.3-16, 2.3-43 to 2.3-46, B.5-7, p. 2 of Table 2.4-1 of submittal]

The adequacy of fire water vessel injection was evaluated through MAAP calculations that account for pump developed head, system flow losses, and backpressure inside the vessel or drywell. If the backpressure were to exceed approximately 120 psig, the fire water system would be unable to provide vessel makeup. [pp. 23, 37 of RAI Responses, pp. 2.3-15, 2.3-16, 2.3-43 to 2.3-46, B.5-7 of submittal]

[3] The fire water system can also be used to provide vessel injection via hose connections to the feedwater system. However, it appears that the IPE did not take credit for this mode of fire water cooling. [pp. 2.3-15, 2.3-16 of submittal]

Credit was taken for vessel makeup with the control rod drive (CRD) pumps, but only for an inadvertent open relief valve transient or a loss of feedwater and stuck open relief valve. CRD at full flow (both pumps) was assumed to be adequate in these cases, but only if other injection systems had previously operated for at least one hour. [pp. 2.3-41, B.5-8, C.1-4 of submittal]

The IPE credited the possibility of successful mitigation of ISLOCA events via injection from (1) the condensate system, with makeup to the condenser hotwell from the condensate storage tank (CST), or (2) the fire water system. The IPE did not credit LPCI or core spray in ISLOCA scenarios due to the adverse environmental conditions expected inside the reactor building. The IPE credited LPCI and fire water as injection sources for large LOCA outside containment events. [pp. 2.3-43, C3.13, C3.14, Figure C.3-6, C.3-7 of submittal]

The RCIC turbine exhaust trip setpoint is set at 46 psig. Because of this relatively high trip setpoint, the IPE considered the possibility of RCIC operation during a loss of containment cooling event. MAAP models accounted for this trip setpoint, as well as RCIC net positive suction head (NPSH) requirements to determine RCIC availability. [p. 16 of RAI Responses]

The Pilgrim emergency operating procedures (EOPs) require that all injection from outside containment be terminated when the torus bottom pressure cannot be maintained below 60 psig. The action precludes any further increase in primary containment water level and is authorized because the consequences of not doing so may cause a loss of primary containment integrity. The EOPs are based on a philosophy that preferentially chooses to maintain primary containment integrity in order to protect against the uncontrolled release of radioactivity. [pp. 16, 17 of RAI Responses]

The existing logic models assumed no core damage if injection systems are successful and containment pressure is successfully controlled. No credit was taken for core cooling after containment failure. The MAAP models used to support the IPE failed injection systems when NPSH limits for pumps were exceeded, or when suppression pool temperatures exceeded limits for HPCI and RCIC pump oil cooling. Pilgrim has a hardened containment vent, and it appears that venting was credited in the front-end analysis as a means of containment pressure control. [pp. 1, 17, 18 of RAI Responses, p. B.6-10 of submittal]

An operator error associated with failure to inhibit the automatic depressurization system (ADS) was not modeled in the ATWS event trees. The licensee states that inclusion of this event would unnecessarily complicate an already complex ATWS event tree, and that no core damage sequences were missed as a result. General Electric analyses [GE TRAGG] were used to show that uncontrolled injection with low pressure systems during an ATWS will not result in substantial fuel damage or threaten the integrity of the reactor vessel. The GE analyses were not available for our review and thus we cannot pass judgment on their validity. However, we do note that failure of operators to inhibit ADS is a relatively small CDF contributor in many other BWR IPE studies.

Pilgrim has two safety valves and four safety relief valves (SRVs). All four relief valves can be automatically opened by the ADS. The number of safety valves and SRVs is less than for comparable plants. The licensee states that the capacity of these valves is also less than for comparable plants. The IPE required that 3 of the 4 SRVs open to provide successful vessel depressurization to allow use of low pressure injection systems during a transient, small LOCA, or medium LOCA. For ATWS events, at least 5 of the 6

safety valves/SRVs were assumed to be required for pressure relief. [pp. 2.2-7, 2.3-11, B.4-1, C.5-16, Table 2.3-1 of submittal]

2.2.3 Systems Analysis.

Systems descriptions are included in Section 2.3 and Appendix B of the submittal. The system descriptions provide information related to system function, system design and operation, and key modeling assumptions. The system descriptions also contain simplified schematics that show major equipment items and important flow and configuration information. A total of 15 systems are described, including ECCS, electrical power, cooling water, instrument air, and HVAC.

2.2.4 System Dependencies.

The IPE addressed and considered the following types of dependencies: motive power, direct equipment cooing, HVAC, and instrumentation and control. Dependency matrices are provided in Tables 2.4-1 through 2.4-4 of the submittal. These tables display, respectively, the following dependency relationships: [Section 2.4 of submittal]

- Frontline support system dependencies
- Support system dependencies
- Initiating event impact on front-line systems
- Frontline system dependencies on other frontline systems.

Room heatup analyses were made to identify equipment HVAC requirements.
The IPE modeled HVAC as a required dependency for RHR and core spray systems. In addition, HVAC was modeled as a dependency for electrical switchgear and battery rooms, though operator compensatory actions were apparently credited for establishment of alternate ventilation in the switchgear rooms. [pp. 2.2-3, 2.2-4, 2.3-38, A1-22 of submittal].

It appears that the IPE has properly accounted for all component and system dependencies.

2.3 Quantitative Process

This section of the report summarizes our review of the process by which the IPE quantified core damage accident sequences. It also summarizes our review of the data base, including consideration given to plant-specific data, in the IPE. The uncertainty and/or sensitivity analyses that were performed were also reviewed.

2.3.1 Quantification of Accident Sequence Frequencies.

The IPE used the small event tree/large fault-tree technique with fault tree linking to quantify core damage sequences. Functional event trees are used. The Cutset and Fault Tree Analysis (CAFTA) software was used to developed the fault trees and perform the accident sequence quantification. Accident sequence cut sets were developed to the level of specific failures or basic events. The accident sequence cut set truncation limit was 1E-09/yr. [pp. 2.1-10 to 2.1-15, 2.2-2, 3.3-1 of submittal]

Non-recovery data for LOSP initiating events were based on the licensee's analysis (lognormal distribution) of plant-specific experience with recovery of total LOSP. Plant experience with total LOSP events is summarized below in Table 2-1. [pp. 8 to 10 of RAI Responses, NSAC-147]

Table 2-1. Plant Experience With Recovery of Loss of Total Offsite Power

Event Date	Cause of Total LOSP	Recovery Time	Used to Support IPE Analysis?	Notes
5-10-77	Snowstorm	2 hours, 40 minutes	Yes	
2-6-78	Severe winds, heavy snow; insulators covered with ice and salt	2 hours, 7 minutes	Yes	
11-19-86	Severe storm	0 hours, 3 minutes	Yes	
11-12-87	Severe winds, heavy wet snow; snow, ice and salt spray on switchyard	11 hours, 0 minutes	No	Incident not considered in IPE because plant had been in cold shutdown for over a year, and the 23 Kv power source had been removed for a <u>once-in-a lifetime</u> plant enhancement (installation of station blackout diesel generator)
10-30-91	Not provided	1 hour, 49 minutes	Yes	

Table 2-2 compares the IPE LOSP non-recovery data with average industry experience reported in an Electric Power Research Institute (EPRI) document [NSAC-147]. [p. 9 of RAI Responses]

Table 2-2. Comparison of IPE and Average Industry LOSP Non-Recovery Data

Time After LOSP Initiator (Hours)	Cumulative Non-Recovery Probability	
	1995 IPE	Average Industry Data (NSAC-147)
2	0.68	0.3
5	0.43	~0.13
12	0.15	~0.06
15	0.094	~0.003
24	0.024	<0.002

As shown in Table 2-2, the IPE non-recovery data are about a factor of 3 higher than average industry experience at and beyond 5 hours. Given Pilgrim's history with severe weather, it is appropriate that the IPE non-recovery factors are less optimistic than average industry experience.

The IPE also took credit for the recovery of failed diesel generators. At 2 hours, the cumulative non-recovery probability is 0.90, while at 12 hours the cumulative non-recovery probability is 0.44[4]. It appears that these diesel generator non-recovery data are based on Pilgrim plant experience. [Figure C2.8, C.2-21 of submittal]

2.3.2 <u>Point Estimates and Uncertainty/Sensitivity Analyses</u>.

The IPE used point value estimates for hardware failures in cases where at least one actual plant failure occurred. For cases where plant-specific experience indicated zero failures, hardware events were quantified either from (1) point estimates using an assumed value of 0.5 for the number of failures over the exposure period, or (2) mean value generic failure data. As discussed in Subsection 2.3.3 of this report, the choice of data in the zero failure case was based on the lower of the plant-specific calculation or generic data. Where plant-specific maintenance unavailability data were available, the IPE used point estimates. Generic maintenance unavailability data used in the analysis represent mean values. The human error probabilities (HEPs) appear to represent point estimates. [pp. A-34, A-35, A-38, A-40 to A-46 of submittal]

The transient initiating events are point-estimates generally derived from plant experience. It appears that point estimates were also used to quantify initiating events for LOCAs, special initiators, and internal flooding events. [Tables 3.1-1, 3.1-3 of submittal]

No statistical uncertainty analysis was performed on the original IPE results that are reported in the submittal. It is not known if a statistical uncertainty analysis was performed in conjunction with the updated 1995 IPE.

Two types of sensitivity analysis were reported by the licensee. In one of these sensitivity analyses, an evaluation of the impact of revised loss of power initiating event frequencies was made on the original IPE model, without consideration of other performance updates and enhancements developed over the last three years. The original IPE used frequencies of .0.475/yr and 0.142/yr, respectively, for loss of preferred (345 Kv) and total LOSP. Using a revised frequency of 0.643/yr for loss of preferred power, the CDF in the original IPE increased by about 8.5% (from 5.85E-05/yr to 6.35E-05/yr). Using a revised frequency of 0.135/yr for total LOSP, the original IPE CDF decreased by no more than about 1%. [p. 12 of RAI Responses]

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[4] It is not clear whether these data pertain to non-recovery of an individual diesel generator or the non-recovery of at least one diesel generator among all three failed diesel generators.

The other type of sensitivity analysis reported by the licensee was the identification of sequences that would exceed 1E-06/yr if HEP error rates were set equal to 1.0[5]. This sensitivity analysis was reported in the submittal for the original IPE. The RAI responses do not indicate whether a similar sensitivity analysis was performed for the 1995 IPE. [pp. 3.3-4, 3.3-13 of submittal]

The submittal presents the results of a Fussel-Vesely importance analysis of important common cause hardware failures and human errors. It is not known whether a revised set of importance measures was also generated for the 1995 IPE. [pp. 3.3-7 to 3.3-12 of submittal]

2.3.3 Use of Plant-Specific Data.

Plant-specific data were used where possible for component failure rates and test/maintenance unavailabilities. Component unavailability estimates were derived for the period between 01/01/81 and 09/30/89 with one major exception. The data used to quantify the HPCI and RCIC systems came from a five year moving average data base for the period between 03/31/87 and 03/31/92. These two systems were quantified with the more recent data to reflect their improved availability since 1990. [p. 26 of RAI Responses, pp. 2.2-2, 2.5-2. 2.5-3, A-2 to A-4 of submittal]

The data collection period of 01/01/81 to 09/30/89 was used to ensure the equivalent of 5 full years of plant operation. Pilgrim was shutdown for most of 1986 and all of 1987 and 1988 for reliability and safety enhancements. [pp. 2.5-2, A-1, A-2 of submittal]

Plant-specific component failure data were used as actual failure rates in the IPE (no update of generic data) if plant-specific experience indicated at least one failure. In cases where plant-specific experience indicated zero failures, the lower of plant-specific or generic data were used. The plant-specific failure rate for a zero failure case was calculated using an assumed value of 0.5 for the number of failures over the exposure period[6]. [p. 2.5-3 of submittal]

Table 2-3 of this review compares Pilgrim plant-specific failure data for selected components to values typically used in PRA and IPE studies, using NUREG/CR-4550 data for comparison. [p. 13 of RAI Responses, pp. A-4 to A-7, A-34, A-35 of submittal]

Table 2-1 indicates that some IPE data for the following component failure modes are a factor of 5 to 10 lower than the generic NUREG/CR-4550 data: HPCI pump (start), RHR pump (start), SSW pump (run), RBCCW pump (start/run), circuit breaker (open), diesel generator (start/run). On the other hand, IPE data for battery chargers and SSW pump start failures are about a factor of 20 higher than the generic data. Other IPE component failure data listed in Table 2-2 are generally within a factor of 3 of corresponding generic data.

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[5] NUREG-1335 requested that licensees identify sequences that have dropped below the reporting criteria because their frequencies have been reduced more than an order of magnitude for credit taken from human recovery actions.

[6] The selection of the lower of these two types of estimates may appear to be overly optimistic. However, a Bayesian analysis using the generic estimate as a prior and with no plant-specific failures would give an estimate below the generic value.

Each diesel generator has a belt-driven fuel oil booster pump to pump fuel into the diesel fuel racks. Although a standby DC-powered fuel oil booster pump is available to supply fuel oil should the belt-driven pump become inoperable, the DC pump has a 20 second start time delay, and therefore is not available to supply fuel on a diesel start. Provided the belt-driven pump remains operable during the first 20 seconds of a diesel run, the DC pump would be able to provide the backup function upon loss of the belt-driven pump. While the plant has experienced failures of this belt-driven system, these failures do not appear to have adversely influenced the diesel generator start/run failure data used in the IPE. The licensee had planned to replace the belt-driven pumps with gear-driven pumps during refueling outage 8 (April-June 1991). However, this planned modification did not take place during this period. The current status of this project is not known. [pp. A-68 to A-70 of submittal] [NRC Memo]

As previously noted, failure data for the HPCI and RCIC systems came from a five year moving average data base for the period between 03/31/87 and 03/31/92. These two systems were quantified with the more recent data to reflect their improved availability since 1990. We have attempted to estimate the level of improvement in the HPCI and RCIC, using data provided in the submittal. The results of our assessment are displayed in Table 2-4. [pp. A-28, A-34, A-47 to A-54 of submittal]

Table 2-3. Plant-Specific Component Failure Data[1]

Component	Failure Mode	Number of Plant Failures	Number of Demands or Hours	IPE Estimate	NUREG/CR 4550 Mean Value Estimate
HPCI Pump	Start	0	76	6.49E-03	3E-02
	Run	0	30.3	5E-03	5E-03
RCIC Pump	Start	3	89	3.37E-02	3E-02
	Run	0	57	5E-03	5E-03
RHR Pump	Start	0	1525	3.28E-04	3E-03
	Run	0	11093.1	3E-05	3E-05
SSW Pump	Start	7	635	1.10E-02	3E-03
	Run	1	164582.1	6.08E-06	3E-05
RBCCW pump	Start	0	666	7.51E-04	3E-03
	Run	0	115572	4.33E-06	3E-05
Diesel-driven fire pump	Start	2	1821	4.86E-03	3E-02
	Run	0	306.51	8E-04	8E-04
HPCI MOV 2301-3 (see note 2)	Open (demand)	1	77	1.30E-02	3E-03
MOV (see note 3)	Open/close (demand)	27	16330	1.65E-03	3E-03
Check valve (all except SSW)	Open/close (demand)	0	1852	2.70E-04	1E-04 (open) 1E-03 (close)
Check valve (SSW)	Open/close (demand)	1	635	1.57E-03	1E-04 (open) 1E-03 (close)
Battery charger	Operate	3	153000	1.95E-05	1E-06
Circuit breaker	Open (demand)	0	2258	2.21E-04	3E-03
	Close (demand)	2	2258	8.86E-04	3E-03
Diesel generator	Start	4	823	4.86E-03	3E-02
	Run	0	1413	3.54E-04	2E-03

Notes: (1) Failures to start, open, close, operate, or transfer are probabilities of failure on demand. The other failures represent frequencies expressed per hour. (2) MOV 2301-3 is the HPCI pump steam admission valve; this valve is subject to significantly higher stresses than the general population of MOVs and is located in an area where maintenance access is difficult (3) Excludes HPCI steam admission valve (2301-3), core spray full flow test valves (1400 A, B), RHR inboard shutdown cooling isolation valve (1001-50), and main steam drain valves (1-220-1, 2).

Table 2-4. HPCI and RCIC Plant Failure Data Before and After Improvement Program

Component Failure Mode	01/01/81 to 09/30/89			03/31/87 to 3/31/92		
	Failures	Demands	Failure Prob.	Failures	Demands	Failure Prob.
HPCI pump- fail to start	10	101	9.9E-02	0	76	6.5E-03 (see note 1)
HPCI MOV- fail to open	3	107	2.8E-02	1	77	1.3E-02
RCIC pump - fail to start	2	190	1.1E-02	3	89	3.4E-02

Notes: (1) Probability calculated based on 0.5 failures.

The data in Table 2-4 clearly indicate an improvement with regard to HPCI pump start failures. The reliability of the RCIC pump appears to have decreased during the most recent data collection period.

As previously discussed in Section 2.2.1 of this report, plant-specific data were used to support the quantification of initiating event frequencies.

2.3.4 Use of Generic Data.

The primary source of generic component unavailability data was NUREG/CR-4550. Additional generic component failure rates were extracted from IEEE-500 and the GE Technical Specification Improvement Analysis for BWR RPS [TS Improv]. [pp. 2.5-3, 2.5-4 of submittal]

Generic component failure data were used in cases where no plant-specific data were available. Also, as previously noted in Subsection 2.3.3 of this report, the lower of plant-specific or generic data were used in cases where plant-specific data indicated zero failures for a component. The plant-specific failure rate for a zero failure case was calculated using an assumed value of 0.5 for the number of failures over the exposure period[7]. [p. 2.5-3 of submittal]

Generic maintenance unavailabilities were used when plant-specific component maintenance unavailabilities were not available. For components for which neither plant-specific nor generic maintenance unavailability data were unavailable, maintenance unavailability was estimated using the following: [p. 2.5-4 of submittal]

- An unavailability of 3E-03 for each loop of a standby safety system, or for a standby train of a system important for continued plant operation
- An unavailability of 5.0E-04 for any major component of such systems, for cases in which it is necessary to break down the maintenance unavailability to a lower level than for a complete loop.

Table A-15 of the submittal lists the NUREG/CR-4550 data that were used in the IPE. From inspection of this table, it appears that NUREG/CR-4550 data were used wherever possible. The component types listed

[7] The selection of the lower of these two types of estimates may appear to be overly optimistic. However, a Bayesian analysis using the generic estimate as a prior and with no plant-specific failures would give an estimate below the generic value.

in submittal Table A-15 include: air-operated valve (AOV), motor-operated valve (MOV), solenoid operated valve, hydraulic operated valve, explosive operated vale, check valve, safety relief valve, mo*r* driven pump, turbine-driven pump, diesel-driven pump, heat exchanger, diesel generator (T/M unavail), elect*ric*al bus, circuit breaker, transformer, strainer, transmitter, and HVAC fan. [A-11, A-40 to A-46 of submittal]

As previously noted in Section 2.2.1 of this report, generic data were used to support the development of certain initiating events.

2.3.5 Common-Cause Quantification.

The estimation of common-cause failure probabilities was based on the MGL method. The primary sources of MGL data appear to be the following: NUREG/CR-2098, NUREG/CR-2099, NUREG/CR-2770, EGG-EA-5623, and EPRI 3967. For components not included in these publications, a set of generic MGL parameters (presented later) was used. A variety of component types were modeled in the common cause analysis, including valves, circuit breakers, batteries, check valves, various pumps, diesel generators, ventilation fans, and temperature switches. [pp. 2.5-4, 2.5-5, 3.3-2, 3.3-3, 3.3-7 to 3.3-10 A-10 to A-12, A-111 to A-143 of submittal]

We performed a comparison of IPE common-cause data with generic beta factors used in the NUREG/CR-4550 methodology document. For component groups with more than two components, the IPE common cause data were used to derive equivalent fractional failures to correspond to the beta factors presented in NUREG/CR-4550[6]. The common cause data comparison is summarized in Table 2-5.

Table 2-5. Comparison of Common-Cause Failure Factors

Component	Failure Mode	Group Size	Equivalent IPE Beta or Fractional Factor	NUREG/CR 4550 Mean Value Beta Factor
Salt Service Water Pump	Start	2	0.024	0.026
	Run	2	0.1	-
RBCCW Pump	Start	2	0.019	0.026
Core Spray Pump	Start	2	0.13	0.15
MOV	Open (demand)	2	0.079	0.088
Diesel Generator	Start	3	0.0016	0.018
	Run	2	0.031	-
	Run	3	0.0016	-

[6] Some of the common cause parameters used in the IPE were extracted from Attachment A.14 of the submittal. Other IPE common cause factors were extracted by inspecting the quantified common cause events listed in submittal Table 3.3-1 and comparing these data to random equipment failure data provided in submittal Table A-10. [pp. 3.3-2, 3.3-3, 3.3-7 to 3.3-10, A-34, A-35, A-111 to A-143 of submittal]

With the exception of the diesel generators, the IPE and NUREG/CR-4550 common cause data listed in Table 2-3 are consistent. The IPE common cause data for the start failure of 3 diesel generators -; an order of magnitude lower than the NUREG/CR-4550 generic data. It is not clear how the licensee quantified the diesel generator common cause data. However, we do note that one of the three diesel generators is a special station blackout diesel generator that was assumed by the licensee to be relatively independent of the other two diesel generators (and presumably would be less likely to fail from a common cause event). In particular, the station blackout diesel generator must be manually started, has independent support systems (including a separate fuel oil supply), and is housed in a separate enclosure. [pp. B.10-4, B.10-5, C.2-3 of submittal]

Table 2-6 below lists the MGL parameters that were used for components where published data were unavailable. The licensee derived the parameters in this table from engineering judgment. These parameters appear to be comparable to common cause data used in other typical IPE/PRA studies. [pp. A-10, A-11 of submittal]

Table 2-6. MGL Parameters Used If Published Data Unavailable

Parameter	Failure to Operate or Actuate	Failure to Continue Functioning or Spurious Operation
Beta	0.1	0.05
Gamma	0.5	0.5
Delta	0.9	0.9
All Others	1.0	1.0

Finally, the IPE modeled common cause failures between HPCI and RCIC pumps and MOVs. A beta factor of 0.018 was used to model common cause events involving the HPCI and RCIC pump start and run functions. It appears that the start and run random failure probabilities used to support the common cause event quantification represent average failure rates of the HPCI and RCIC pumps. [pp. A-112, A-113 of submittal]

2.4 Interface Issues

This section of the report summarizes our review of the interfaces between the front-end and back-end analyses, and the interfaces between the front-end and human factors analyses. The focus of the review was on significant interfaces that affect the ability to prevent core damage.

2.4.1 Front-End and Back-End Interfaces.

The IPE credited the possibility of successful mitigation of ISLOCA events via injection from (1) the condensate system, with makeup to the condenser hotwell from the condensate storage tank (CST), or (2) the fire water system. The IPE did not credit LPCI or core spray in ISLOCA scenarios due to the adverse environmental conditions expected inside the reactor building. The IPE credited LPCI and fire water as injection sources for large LOCA outside containment events. [pp. 2.3-43, C3.13, C3.14, Figure C.3-6, C.3-7 of submittal]

The RCIC turbine exhaust trip setpoint is set at 46 psig. Because of this relatively high trip setpoint, the IPE considered the possibility of RCIC operation during a loss of containment cooling event. MAAP models accounted for this trip setpoint, as well as RCIC net positive suction head (NPSH) requirements to determine RCIC availability. [p. 16 of RAI Responses]

The Pilgrim emergency operating procedures (EOPs) require that all injection from outside containment be terminated when the torus bottom pressure cannot be maintained below 60 psig. The action precludes any further increase in primary containment water level and is authorized because the consequences of not doing so may cause a loss of primary containment integrity. The EOPs are based on a philosophy that preferentially chooses to maintain primary containment integrity in order to protect against the uncontrolled release of radioactivity. [pp. 16, 17 of RAI Responses]

The existing logic models assumed no core damage if injection systems are successful and containment pressure is successfully controlled. No credit was taken for core cooling after containment failure. The MAAP models used to support the IPE failed injection systems when NPSH limits for pumps were exceeded, or when suppression pool temperatures exceeded limits for HPCI and RCIC pump oil cooling. Pilgrim has a hardened containment vent, and it appears that venting was credited in the front-end analysis as a means of containment pressure control. [pp. 1, 17, 18 of RAI Responses]

The front and back-end analyses were coupled by linking core damage cut sets directly into containment system event trees (CSETs). The output of the CSETs was used to generate a set of plant damage states (PDSs). The PDSs were subsequently evaluated by containment phenomenological event trees (CPETs) to generate source term frequencies. The process used to couple the front- and back-end analysis appears to be comparable with similar processes used in other PRA/IPE studies. [p. 49 of RAI Responses, pp. 4.0-2, 4.3-14 of submittal]

2.4.2 Human Factors Interfaces.

Based on our review of the front-end analysis, the operator actions were found to be important: [pp. 3.4-22 to 3.4-25]

- Operator actions needed for reactor depressurization
- Operator actions needed for use vessel injection via firewater
- Operator actions needed for SLC injection

Table 3.3-2 of the submittal provides importance measures for human events pertinent to the original IPE analysis. A comparable set of importance measures relevant to the updated (1995) IPE was not provided. The licensee states that credit was taken only for proceduralized human actions. [pp. 2.2-4, 3.3-11, 3.3-12 of submittal]

The IPE credited use of the fire water system for alternative vessel injection during transient events, small LOCAs, and reference line breaks. This action, which is proceduralized, requires installation of a spool piece to connect the fire water system to RHR system discharge piping. The spool piece is located in a cabinet immediately adjacent to its installation point, and can be connected without special tools due to the use of quick snap couplings. The licensee states that operators are trained and tested in the use of the fire water

system. In addition to installation of the spool piece, use of the fire water system requires that operators open two valves. The availability of DC power is also required to allow vessel depressurization via the safety relief valves (SRVs). [pp. 23, 34, 37 of RAI Responses, pp. 2.3-15, 2.3-16, 2.3-43 - 2.3-46, B.5-7, p. 2 of Table 2.4-1 of submittal]

General Electric analyses were used to show that uncontrolled injection with low pressure systems during an ATWS will not result in substantial fuel damage or threaten the integrity of the reactor vessel. Consequently, an operator error associated with failure to inhibit the automatic depressurization system (ADS) was not modeled in the ATWS event trees. [p. 41 of RAI Responses, pp. C.5-7, C.5-8 of submittal]

The IPE took credit for the recovery of failed diesel generators. At 2 hours, the cumulative non-recovery probability is 0.90, while at 12 hours the cumulative non-recovery probability is 0.44[9]. It appears that these diesel generator non-recovery data are based on Pilgrim plant experience. [Figure C2.8, C.2-21 of submittal]

Pilgrim has a hardened containment vent, and it appears that venting was credited in the front-end analysis as a means of containment pressure control. [pp. 1, 17, 18 of RAI Responses, p. 3.3-11 of submittal]

2.5 Evaluation of Decay Heat Removal and Other Safety Issues

This section of the report summarizes our review of the evaluation of Decay Heat Removal (DHR) provided in the submittal. Other GSI/USIs, if they were addressed in the submittal, were also reviewed.

2.5.1 Examination of DHR.

The licensee specifically addresses DHR and its contribution to CDF. The IPE DHR analysis was based on a narrow DHR definition, namely removal of decay heat from containment. This definition does not address core cooling aspects of DHR. In order to resolve USI A-45, licensees were requested to examine DHR for its capability during both core cooling and containment heat removal phases and for all accidents except large LOCAs, ATWS events, and ISLOCAs.

Using qualitative discussions, the licensee concluded that the containment DHR reliability is high due to the availability of multiple containment systems, specifically: (1) main condenser, (2) RHR in suppression pool cooling, wetwell spray, drywell spray, or shutdown cooling mode, (3) reactor water cleanup system (RWCU) either through the non-regenerative heat exchanger or in a feed and bleed mode to the main condenser, (4) containment venting through the standby gas treatment system, and (5) direct torus hardened vent. Even if all these containment DHR systems were to fail, a significant amount of time is available for repair and recovery efforts. Specifically, 34 hours would be available before containment design limits are exceeded, and 47 hours would be available before the ultimate containment capacity is reached. The licensee notes that the IPE did not take credit for heat removal via the RWCU system. No DHR-related vulnerabilities were identified. [pp. 3.6-2, 3.6-3, 3.6-13, 3.6-14 of submittal]

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[9] It is not clear whether these data pertain to non-recovery of an individual diesel generator or the non-recovery of at least one diesel generator among all three failed diesel generators.

While the IPE DHR analysis was limited to removal of decay heat from containment, the licensee does identify the IPE accident classes that would pertain to the expanded (A-45) definition of DHR. These accident classes are listed below: [p. 25 of RAI Responses]

- Class IA: Failure of high pressure makeup and failure to depressurize (TQUX)
- Class IB: Station blackout
- Class ID: Loss of both high and low pressure coolant injection (TQUV)
- Class II: Failure of DHR from containment (TW)
- Class IIIB: Small or medium LOCAs for which reactor vessel cannot be depressurized (S1QUX or S2QUX)
- Class IIIC: Medium LOCAs for which there is inadequate vessel makeup (S1V)

2.5.2 Diverse Means of DHR.

The IPE evaluated the diverse means for accomplishing DHR, including use of: the power conversion system, RCIC, HPCI, and vessel depressurization to allow injection with low pressure systems (including fire water). Pilgrim has a hardened containment vent that appears to have been credited in the IPE.

2.5.3 Unique Features of DHR.

The unique features at Pilgrim that directly impact the ability to provide DHR are as follows:

- Ability to perform vessel injection with fire water system. Alternate vessel injection can be accomplished with the fire water system. Because one of the fire water pumps is diesel-driven, this method of injection can be used during station blackout. This design feature tends to lower the CDF. [pp. 2.2-6 of submittal]

- Motor-driven feedwater pumps. Because the feedwater pumps are motor-driven instead of turbine-driven, they are not disabled in MSIV closure events and may remain available for vessel injection. This design feature tends to lower the CDF. [p. 2.2-6 of submittal]

- Limited vessel pressure relief capability. Pilgrim has a more limited vessel pressure relief capability than other BWRs of similar design. This limited pressure relief capability is due to the relatively small number of relief valves (2 code safety valves and 4 safety relief valves) and the relatively small capacity of each individual valve. This design feature tends to increase the CDF.

- Hardened torus vent. The availability of a hardened torus vent provides an additional means of providing containment pressure control and decay heat removal. This design feature tends to lower the CDF. [pp.2.2-6, 3.6-12, B.6-8, B.6-9 of submittal]

2.5.4 Other GSI/USIs Addressed in the Submittal.

No GSI/USIs other than A-45 are addressed by the IPE. [p. vii of transmittal letter]

2.6 Internal Flooding

This section of the report summarizes our reviews of the process used to model internal flo··ng and of the results of the analysis of internal flooding.

2.6.1 Internal Flooding Methodology.

The internal flooding analysis was based in part on previous flood-related studies of the Pilgrim plant, including analyses included in the Pilgrim UFSAR. Plant walkdowns were also used to support the analysis. Postulated flooding events were considered to be bounded by pipe breaks of high volumetric flow rates in terms of their impact on plant systems, therefore events such as over-filling water tanks, hose ruptures, and pump seal leaks were not considered. Fire protection system breaks were included in the analysis. The only safety-related areas in which fire protection system flooding is the predominate source are the switchgear rooms. Wire mesh panels/doors prevent any accumulation in these areas. [pp. 2.1-6, C.4-17 to C.4-23 of submittal]

The IPE explicitly considered equipment failure from submergence, though effects from spray were not considered. The licensee states that equipment damage due to flooding envelopes spray-induced equipment damage. The licensee further states that spray-induced effects were addressed in the Pilgrim IPE External Events (IPEEE) study submitted to NRC in 1994. The IPEEE study concluded that plant design features preclude spray-related effects on equipment. For example, reactor building elevations 23' and 51' have berms and ramps to contain water from a water curtain, thereby protecting important equipment. These areas also have equipment spray shields. In addition, some equipment items are surrounded by spray guards in instances where the equipment items are in close proximity to pipes (especially high energy lines). Spray guards are also used in situations where equipment items are located near or under fire sprinklers. [pp. 5, 6 of RAI Responses, p. C.4-19 of submittal]

The flooding analysis methodology included consideration of the following: [p. C.4-23 of submittal]

- Identification of potential flood locations
- Determination of blowdown/spillage volumes
- Determination of spaces affected by each flooding event
- Determination of area of affected spaces
- Calculation of flood levels (flood level = volume/area)

Quantification of flood scenarios appears to have been based on the Level 1 logic models. It is not clear whether credit was taken for operator mitigating actions.

2.6.2 Internal Flooding Results.

The IPE analysis reported in the submittal (pre-1995 IPE) identified only one initiating event that could also disable potential mitigating systems useful in responding to the event. This initiating event involves a feedwater break inside the steam tunnel that disables the feedwater system and also submerges CRD components. This scenario has a CDF contribution of 2.27E-07/yr. [p. 6 of RAI Responses, pp. C.4-19, C.4-20, C.4-22, C.4-23 of submittal]

Result from the 1995 IPE indicate that the total internal flooding CDF contribution is 6.1E-08/yr. The sequence(s) associated with the 1995 IPE are not provided. [p. 22 of RAI Responses]

2.7 Core Damage Sequence Results

This section of the report reviews the dominant core damage sequences reported in the submittal. The reporting of core damage sequences- whether systemic or functional- is reviewed for consistency with the screening criteria of NUREG-1335. The definition of vulnerability provided in the submittal is reviewed. Vulnerabilities, enhancements, and plant hardware and procedural modifications, as reported in the submittal, are reviewed.

2.7.1 Dominant Core Damage Sequences.

The CDF point estimate from the 1995 IPE model (including internal flooding) is 2.84E-05/yr[10]. Accident types and their contributions to CDF are provided below in Table 2-7[11]. [pp. 20 to 22 of RAI Responses]

Initiating events and their percent contribution, are listed below in Table 2-8. [p. 22 of RAI Responses]

The 10 most dominant functional core damage sequences are summarized below in Table 2-9 of this report. [pp. 20, 21 of RAI Responses, Appendix C of submittal]

Table 2-7. Accident Types and Their Contribution to Core Damage Frequency

Accident Type	CDF Contribution (per/yr)	Percent Contribution to CDF
Transient	2.0E-05	70
ATWS	4.5E-06	16
LOCA (see note 1)	2.9E-06	10
Station Blackout	9.6E-07	3
ISLOCA	1.0E-07	0.4
Internal Flood	6.1E-08	0.2

Notes: (1) LOCA category includes IORV (6.1E-07/yr), large LOCA outside containment (1.0E-07/yr), and reference line break (2.0E-08/yr).

Table 2-8. Initiating Events and Their Contribution to Core Damage Frequency

[10] A CDF estimate of 5.85E-05/yr was reported in the previous IPE analysis described in the submittal. Much of the difference between the original and revised IPE CDF values is due to elimination of the HPCI room cooling dependency, improvements in HPCI/RCIC reliability data, and more optimistic ADS success criteria. [p. 1 of RAI Responses, p. 3.4-3 of submittal]

[11] The data contained in this table were derived from revised Tables 3.4-1 and 3.4-2 included in the RAI responses. [pp. 20 to 22 of RAI Responses]

Initiating Event	CDF Contribution / yr.	% Cont. to CDF
Partial LOSP (345 Kv)	8.46E-06	30
Manual shutdown	5.26E-06	19
Full LOSP (345 & 23 Kv)	2.78E-06	10
Turbine trip and reactor trip	2.78E-06	10
Loss of feedwater	2.27E-06	8.0
Medium LOCA	1.70E-06	6.0
Loss of condenser vacuum	1.00E-06	3.5
Loss of DC bus B	9.36E-07	3.3
MSIV closure	8.11E-07	2.9
Loss of salt service water (SSW)	6.82E-07	2.4
Inadvertent open relief valve (IORV)	6.14E-07	2.2
Reactor vessel rupture	3.00E-07	1.1
Large LOCA	1.62E-07	0.6
Small LOCA	1.21E-07	0.4
Loss of DC Bus A	1.09E-07	0.4
Main steam line break	1.00E-07	0.4
internal flood	6.07E-08	0.2
Core spray ISLOCA	5.00E-08	0.2
LPCI ISLOCA	5.00E-08	0.2
Reference line break	1.95E-08	0.06
Loss of RBCCW	1.38E-09	0.05

Table 2-9. Top 10 Dominant Functional Core Damage Sequences

Initiating Event	Dominant Subsequent Failures in Sequence	% Contribution to Total CDF
LOSP	Failure of high pressure injection, failure to depressurize	30
Transient	Failure of high pressure injection, failure to depressurize	22
Transient	Loss of DHR (TW sequence)	12
Transient (non-isolation)	RPS mechanical failure, operator failure to inject SLC (ATWS)	6
LOSP	Offsite power not recovered in 24 hours, failure of containment pressure control, failure of reactor inventory control	4
Medium LOCA	Failure of high pressure injection, failure to depressurize	4

LOSP	Failure of battery load shedding, offsite power not recovered in 12 hours, fail to recover diesel generator in 12 hours (station blackout)	3
Transient (non-isolation)	RPS mechanical failure, SRVs fail to open (ATWS)	2
Transient (isolation)	RPS mechanical failure, operator failure to inject SLC (ATWS)	2
Medium LOCA	Failure of low pressure injection after successful high pressure injection	2

The submittal provides the results of a Fussel-Vesely importance analysis of important common cause hardware failures and human errors. However, the RAI responses do not provide an updated set of these importance measures for the 1995 IPE. The most important events based on the Fussel-Vesely measures generated in the original IPE are listed below: [pp. 3.3-7 to 3.3-12 of submittal]

- Operator fails to depressurize (non-ATWS)
- Common cause failure of SRVs 203-3A, B, C, D to open
- Common cause failure of HPCI/RCIC pumps to start
- Common cause failure of 3 of 4 SRVs to open (203-3A, B, C, D, six separate events)

2.7.2 Vulnerabilities.

The licensee used the following criteria to search for vulnerabilities:

- Are there any new or unusual means by which core damage or containment failures occur as compared to those identified in other PRAs?
- Do the results suggest that the Pilgrim core damage frequency would not be able to meet the NRC's safety goal for core damage (1E-04/yr)?

Based on the above criteria, the licensee concluded that there are no vulnerabilities at Pilgrim. [p. 5.0-1 of submittal]

2.7.3 Proposed Improvements and Modifications.

It appears that two plant improvements were suggested as a result of the IPE. One improvement involved a review of procedures for DC bus loss. At the time of the original IPE analysis was performed, procedures directed operators to trip all loads powered by AC buses associated with a failed DC bus. In the event both DC buses were lost, these proceduralized operator actions would have resulted in the loss of all feedwater pumps. The procedures have been revised to allow operator judgment with regard to shedding of AC bus loads following DC bus failures. This improvement was not credited in the original IPE, and it is not clear whether it was credited in the updated (1995) IPE. These procedure revisions would have reduced the CDF reported in the submittal by about 5% (from 5.85E-05/yr to 5.55E-05/yr). The CDF impact on the current 1995 IPE was not provided. [p. 36 of RAI Responses, pp. 3.3-11 of submittal]

The other improvement suggested as a result of the IPE is a procedural change to allow operators to use fire water for drywell sprays. The current status and CDF impact of this procedure change were not provided. [p. 2.1-22 of submittal]

The licensee provided information concerning plant changes made in response to the Station Blackout Rule, and other modifications separate from the Station Blackout Rule that reduce the station blackout CDF. This information is summarized below in Table 2-10. The licensee does not have any analyses related to the CDF impact of these modifications. [pp. 8 to 10, 12, 50, 51 of RAI Responses, p. 3.1-2 of submittal]

Table 2-10. Summary of Plant Changes That Directly Affect Station Blackout

Description of Plant Change	Status	Plant Change Accounted for in IPE?	Notes
Modifications Specifically Related to Station Blackout Rule			
Install third (station blackout) diesel generator	Completed	Yes	
Add pull-to-lock switches for circuit breakers associated with RHR pumps, core spray pumps, and shutdown transformer feeders	Unknown	No	
Install load shedding switches to initiate load shedding logic on AC trains A and B	Unknown	No	
Modifications Separate From Station Blackout Rule			
Application of special coating to 345 Kv switchyard insulators to reduce likelihood of salt buildup/flashover	Completed	Yes	
Phase separation of 345 Kv feeder	Completed	Yes	
Switchyard betterment program to reduce probability of salt buildup	Completed	No	Completed during 1994 and 1995 outages

3. CONTRACTOR OBSERVATIONS AND CONCLUSIONS

This section of the report provides an overall evaluation of the quality of the IPE based on 's review. Strengths and weaknesses of the IPE are summarized. Important assumptions of the model are summarized. Major insights from the IPE are presented.

Strengths of the IPE are as follows: The IPE goes beyond the bounds of some other BWR IPE/PRA studies by considering and modeling common cause failures between the HPCI and RCIC systems.

No major weaknesses of the IPE were identified.

One weakness of the <u>submittal</u> was identified, namely that the licensee's DHR analysis was limited to removal of decay heat from containment. This narrow definition of DHR does not address core cooling aspects of DHR. In order to resolve USI A-45, licensees were requested to examine DHR for its capability during both core cooling and containment heat removal phases and for all accidents except large LOCAs, ATWS events, and ISLOCAs. While the licensee's narrow definition of DHR is judged to be a weakness of the <u>submittal</u>, both core and containment cooling has been accounted for in the overall IPE analysis process. In our judgment, the IPE models are capable of identifying DHR-related vulnerabilities.

Significant findings on the front-end portion of the IPE are as follows:

- The Pilgrim plant is located in a region of the country that is prone to more frequent occurrences of severe weather than many other nuclear plant sites. Consequently, LOSP frequencies and non-recovery probabilities are higher at Pilgrim compared to average industry data. Even so, station blackout contributes only about 3% of the total CDF at Pilgrim. The relatively low CDF contribution of station blackout at Pilgrim is due to: (1) a 14 hour battery capacity (with credit for load shedding), (2) the availability of a station blackout diesel generator, (3) the availability of a separate 23 Kv offsite power source for plant shutdown functions, (4) the availability of an AC-independent source of vessel injection (fire water), and (5) credit for recovery of failed diesel generators.

- ATWS sequences contribute 16% to the total CDF. About 55% of the ATWS contribution is due to operator failure to initiate standby liquid control (SLC) injection. Another 20% of the ATWS contribution is related to inadequate pressure relief caused by failure of sufficient safety valves/safety relief valves to open.

- Common cause failures of safety relief valves (SRVs) are important contributors in sequences where high pressure injection is unavailable and depressurization fails. As previously noted in the discussion on plant features, Pilgrim has a more limited vessel pressure relief capability than other BWRs of similar design.

4. DATA SUMMARY SHEETS

This section of the report provides a summary of information from our review.

<u>Initiating Event Frequencies</u>

Initiating Event	Frequency per year
Turbine and Reactor Trip	2.07
Loss of Feedwater	4.14E-01
Loss of Feedwater (unrecoverable)	1.90E-01
Loss of Condenser Vacuum	5.00E-01
MSIV Closure	4.14E-01
Inadvertent Opening Relief Valve (IORV)	3.00E-01
Loss of Offsite Power (345 Kv + 23 Kv)	1 [illegible]2E-01
Loss of 345 Kv Power Only	4.75E-01
Manual Shutdown	3.89
LOCAs Inside Containment	
Small LOCA	8.0E-03
Medium LOCA	3.0E-03
Large LOCA	7.0E-04
RPV Rupture	1.0E-05
LOCAs Outside Containment	
LPCI Interfacing Systems LOCA	5.0E-07
CS Interfacing Systems LOCA	5.0E-07
Large LOCA	1.4E-07
Internal Floods	8.2E-03
Single DC Bus Failure (50% Recovery)	3.0E-03
Reactor Water Level Reference Line Breaks	4.0E-02
Loss of Service Water	2.7E-04
Loss of Reactor Building Cooling Water	1.9E-04
Loss of Turbine Building Cooling Water	1.4E-03
Loss of Instrument Air	2.0E-04

<u>Overall CDF</u>

The point estimate CDF for Pilgrim is 2.84E-05/yr, including internal flooding. The CDF contribution from flooding is 6.1E-08/yr.

Dominant Initiating Events Contributing to CDF

Partial LOSP	30%	
Manual shutdown		19%
Full LOSP (345 & 23 Kv)		10%
Turbine trip and reactor trip	10%	
Loss of feedwater		8%
Medium LOCA	6%	
Loss of condenser vacuum	4%	
Loss of DC Bus B		3%
MSIV closure	3%	

Dominant Hardware Failures and Operator Errors Contributing to CDF

Dominant hardware failures contributing to CDF include:

Common cause failure of SRVs 203-3A, B, C, D to open
Common cause failure of HPCI/RCIC pumps to start
Common cause failure of 3 of 4 SRVs to open (203-3A, B, C, D, six separate events)

Dominant human errors and recovery factors contributing to CDF include:

Failures to depressurize (non-ATWS)
Failure to inject SLC before heat capacity temperature limit

Dominant Accident Classes Contributing to CDF

Transient		70%
Anticipated Transient Without Scram (ATWS)	16%	
LOCAs	10%	
Station Blackout		3%
ISLOCA	0.4%	
Internal Flood	0.2%	

Design Characteristics Important for CDF

The following design features impact the CDF:

- Fourteen hour battery capacity. With credit for load shedding, the batteries can provide necessary power during station blackout for approximately 14 hours. The 14 hour battery lifetime is longer than battery lifetimes at many other plants. This design feature tends to lower the CDF.

- Station blackout diesel generator. A station blackout diesel generator has been installed at Pilgrim. This design feature tends to lower the CDF.

- Special 23 Kv offsite power line for plant shutdown functions. Pilgrim has a special 23 Kv offsite power connection that can be used to power emergency buses in the event offsite power from the two 345 Kv sources is lost. The 23 Kv power line is routed through a separate switchyard into the station shutdown transformer. Plant experience has shown that the 23 Kv line is more resistant to weather-related effects than the 345 Kv sources. This design feature tends to lower the CDF.

- Ability to perform vessel injection with fire water system. Alternate vessel injection can be accomplished with the fire water system. Because one of the fire water pumps is diesel-driven, this method of injection can be used during station blackout. This design feature tends to lower the CDF.

- Limited vessel pressure relief capability. Pilgrim has a more limited vessel pressure relief capability than other BWRs of similar design. This limited pressure relief capability is due to the relatively small number of relief valves (2 code safety valves and 4 safety relief valves) and the relatively small capacity of each individual valve. This design feature tends to increase the CDF.

- Hardened torus vent. The availability of a hardened torus vent provides an additional means of providing containment pressure control and decay heat removal. This design feature tends to lower the CDF.

- Portable diesel-driven air compressor. A portable diesel-driven air compressor can be manually connected to the compressed air system. This additional source of compressed air tends to reduce the CDF.

- Diverse instrument nitrogen supplies. Diverse means are available for supping nitrogen to support important functions, for example the automatic depressurization system (ADS) valves. Sources of nitrogen include banks of bottled nitrogen and a trailer-mounted set of liquid nitrogen tanks. This design feature tends to reduce the CDF.

- Independence of diesel generators from external cooling water sources. The diesel generators (including the station blackout diesel generator) are self-cooled. This design feature lowers the CDF.

Modifications

It appears that the following two plant improvements were identified as a result of the IPE:

- Modify loss of DC procedures to allow operator judgment for load shedding of AC buses associated with failed DC supplies (completed)
- Modify procedures to allow operators to use fire water for drywell sprays

Other USI/GSIs Addressed

No generic safety issues (GSIs)/USIs other than A-45 are addressed by the IPE.

Significant PRA Findings

Significant findings on the front-end portion of the IPE are as follows:

- The Pilgrim plant is located in a region of the country that is prone to more frequent occurrences of severe weather than many other nuclear plant sites. Consequently, LOSP frequencies and non-recovery probabilities are higher at Pilgrim compared to average industry data. Even so, station blackout contributes only about 3% of the total CDF at Pilgrim. The relatively low CDF contribution of station blackout at Pilgrim is due to: (1) a 14 hour battery capacity (with credit for load shedding), (2) the availability of a station blackout diesel generator, (3) the availability of an AC-independent source of vessel injection (fire water), and (4) credit for recovery of failed diesel generators.

- ATWS sequences contribute 16% to the total CDF. About 55% of the ATWS contribution is due to operator failure to initiate standby liquid control (SLC) injection. Another 20% of the ATWS contribution is related to inadequate pressure relief caused by failure of sufficient safety valves/safety relief valves to open.

- Common cause failures of safety relief valves (SRVs) are important contributors in sequences where high pressure injection is unavailable and depressurization fails. As previously noted in the discussion on plant features, Pilgrim has a more limited vessel pressure relief capability than other BWRs of similar design.

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