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Nine Mile Point 1

Technical Evaluation Report on the Individual Plant Examination Front End Analysis

NRC-04-91-066, Task 30

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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 Nine Mile Point 1. This review is based on information contained in the IPE submittal along with the licensee's responses to Requests for Additional Information (RAI).

E.1 Plant Characterization

The Nine Mile Point 1 plant is co-located with the Nine Mile 2 plant, but shares few systems with unit 2, because the two plants are different vintage Boiling Water Reactors (BWR). The plant is located on the southeast shore of Lake Ontario in New York, seven miles northeast of Oswego, New York. Nine Mile Point 1 is a BWR 2 with a Mark I containment. General Electric was the Nuclear Steam System Supplier (NSSS); the utility, Niagara Mohawk Power Corporation, was the architect engineer (AE) and Stone and Webster was the constructor. The Unit achieved commercial operation in December 1969. The rated power is 1850 megawatts thermal (MWt) and 620 megawatts electric (MWe) net.

Design features at Nine Mile Point 1 that impact the core damage frequency (CDF) are as follows:

- Emergency (isolation) condensers (ECs) The ECs require no electrical power to provide for core cooling, and this tends to decrease CDF. Per design, the ECs alone provide no makeup to the vessel and thus excessive leakage or unisolated seal Loss of Coolant Accidents (LOCAs) cannot be mitigated with the ECs; this tends to increase the CDF.

- Hardened containment vent The hardened containment vent decreases the CDF by providing a backup to loss of containment cooling whereby continued support of core cooling systems can be provided.

- Eight-hour battery lifetime The 8 hour battery lifetime tends to decrease CDF, since it is relatively long compared to battery lifetimes at some other BWRs, thereby allowing for increased likelihood of recovery of offsite power.

- Diesel driven firewater The availability of diesel driven firewater tends to lower the CDF since this source can be used to provide makeup to the ECs and to inject to the vessel via the feedwater piping.

- Ability to power control rod drive (CRD) pumps off 1E power The ability to power the CRD pumps with the diesel generators (DGs) tends to decrease the CDF since this provides an additional source of makeup to the vessel even if offsite power is lost.

E.2 Licensee's IPE Process

The IPE is a level 2 probabilistic risk assessment (PRA). The licensee initiated work on a PRA for Nine Mile Point 1 in response to the Generic Letter 88-20. They modeled the plant as of January, 1993. No pending improvements were credited in the IPE model.

Five full-time engineers from the utility were assigned to the IPE effort. As-needed support was provided from in-line utility organizations, including: operations, nuclear technology, safety analysis, systems engineering, reactor engineering, training, electrical design, fuels, and design basis reconstitution. Consultants from the following organizations were utilized: General Physics, Fauske and Associates, GKA, Haliburton NUS, and EQE.

Three types of walkdowns were performed for the IPE. The first walkdown was a week in duration, performed during shutdown, and focused on the primary containment and equipment within primary containment. The second walkdown was performed at power, and it focused on the reactor building. The third walkdown was a collection of individual walkdowns, performed as necessary in support of modeling of individual systems and in support of the model for internal flooding.

Major documentation used in the IPE included: the Updated Final Safety Analysis Report (UFSAR), procedures, drawings, design basis documents, equipment qualification reports, Licensee Event Reports (LERs), in service test results, and maintenance work reports.

Numerous PRAs were reviewed during the preparation of the IPE for Nine Mile Point 1, especially the IPE for Nine Mile Point 2.

The submittal states that: "continued application of IPE is a priority for NMPC". So far, the IPE has been used by the utility for developing an emergency diesel generator reliability program, supporting operator training, and enhancing operating procedures.

E.3 Front-End Analysis

The methodology chosen for the Nine Mile Point 1 IPE front-end analysis was a Level I PRA; the large event tree/small fault tree technique with support state modeling was used and quantification was performed with the RISKMAN computer code.

The IPE quantified 23 initiating events: 7 LOCAs, 8 plant specific support system failures, and 8 generic transients. The IPE developed 4 large, systemic event trees for frontline systems, to model the plant response to each initiating event. Two large support state event trees were developed.

Loss of heating, ventilating and air conditioning (HVAC) was considered to not be an important initiating event. Loss of instrument air was modeled as an initiating event.

The criterion used for inadequate core cooling was temperature in excess of 2500 F peak core temperature.

Success criteria were developed based on thermal hydraulic analyses, mostly with Modular Accident Analysis Program (MAAP) calculations, the UFSAR, and engineering judgement.

The IPE used plant specific data from 1984 through 1992, for selected components. Generic data were updated with plant specific data for the selected components and for test and maintenance unavailabilities.

The IPE modeled support system dependencies in two large support system fault trees. The fault trees were solved for each core damage sequence by quantifying the frontline event tree sequences conditional on the support states specified by the support state event trees. Tables of inter-system dependencies were included in the submittal. No HVAC systems were considered required. Instrument air was modeled.

The Multiple Greek Letter (MGL) method was used to model common cause failure, and the Pickard, Lowe and Garrick (PL&G) generic common cause failure database was used to quantify common cause failures. Common cause failures were modeled within systems.

The internal flooding analysis was performed in the following manner. Potential flood sources were identified using information on the plant layout combined with a review of industry flooding experience. A plant walkdown was performed to collect additional information, such as the relative locations of equipment in flood zones. Flood propagation was evaluated to determine the extent of equipment damage from flooding events. Flood scenarios were then conservatively quantified using industry data without credit for operator intervention. For those scenarios surviving this conservative analysis, more detailed analyses were performed, including operator intervention. The general transient event tree was used to evaluate core damage from floods that did not in-and-of-themselves directly cause core damage; this evaluation addressed core damage from a flood initiating event with its resultant failure of equipment, combined with subsequent random failures of equipment.

The model for internal flooding does not consider spray induced failure of equipment. The submittal states that an internal flooding analysis performed by the utility in 1972, and reviewed by the NRC in 1974, concluded that vital equipment is adequately protected from spray and splashing failures by the use of drip-proof motors or shields.

The total CDF from internal initiating events is 5.5E-6/year. Internal flooding was screened from quantification for the CDF, based on semi-quantitative evaluations of flood scenarios.

Initiating events that contribute the most to CDF, and their percent contribution, are as follows:

Loss of Offsite Power leading to Station Blackout	63%
Medium LOCA in Water Line	7%
Loss of Instrument Air	7%
Small LOCA in Water Line	4%
Reactor Scram Initiating Events with anticipated transient without scram (ATWS)	3%

A more detailed listing of the CDF by initiating event is provided in Section 2.7.1 of this report.

The CDF grouped by class of accident is as follows:

Station Blackout	64%
Loss of Injection	19%
ATWS	10%
Loss of DHR	6%.

Based on contributions to the CDF, the following systems are most important, listed in decreasing order:

Emergency AC Power
Emergency DC Power
Relief Valves Reclosing
Diesel Firewater Injection to Vessel.

Based on contributions to CDF, the following operator actions are most important, listed in decreasing order:

AC power recovery
load shedding DG under LOCA conditions
depressurization
preventing EC isolation and EC recovery after isolation
calibration of core spray injection permissive
feedwater control given loss of instrument air
DC load shedding given station blackout
alignment of torus cooling mode of containment spray.

Each front end core damage accident sequence was linked directly to the level 2 containment tree, so binning of core damage sequences into Plant Damage States (PDSs) was not required. However, the submittal does provide a binning classification for core damage sequences to facilitate the assessment of results. These bins are referred to as Core Damage End States, and are tabulated in the submittal.

Based on our review, the following modeling assumptions are somewhat unique and can have a significant impact on the overall CDF:

(a) assignment of 0.05 for the probability of a seal LOCA with loss of cooling to the recirculation pump seals
(b) the assumption that no HVAC or ventilation systems are required
(c) the assumption that the core spray pumps, rated for 140 deg. F, can survive up to over 300 deg. F with a 0.5 probability.

All of these assumptions tend to lower the CDF. The first assumption makes a seal LOCA relatively unlikely. The second assumption means that all HVAC and ventilation systems can fail and not impact the ability to prevent core damage. The third assumption decreases the likelihood of loss of core cooling given loss of containment heat removal.

E.4 Generic Issues

The IPE specifically addressed loss of Decay Heat Removal (DHR). The IPE considers DHR to be the final heat sink. The contribution of loss of DHR to the CDF is 6%. This relatively low contribution is due to the variety of methods available to provide DHR at Nine Mile Point 1, including:

power conversion system
emergency condensers
containment spray system
shutdown cooling system
containment venting.

No vulnerabilities associated with loss of DHR were identified by the IPE.

The submittal addresses three NRC questions from Generic Letter 91-06 related to resolution of GI A-30, "Adequacy of Safety Related DC Power Supplies".] The licensee requests that GL 91-06 be resolved based on the IPE submittal.

E.5 Vulnerabilities and Plant Improvements

The licensee does not define vulnerability. The submittal concludes: "With regard to core damage frequency, there are no unusual or unique contributors to core damage that suggest a plant specific vulnerability."

The submittal discusses improvements judged to have the most potential benefit for reducing risk. These potential improvements are as follows.

Load shedding of non-1E battery loads so the non-1E battery system can be used for supply of DC power after the 1E batteries fail at 8 hours during station blackout

Use of a portable battery charger

Improved calibration of low vessel pressure emergency core cooling system (ECCS) permissive sensors

Use of plant specific data exclusively for failure of a relief valve to reclose instead of generic data updated with plant specific data, since failure to reclose has never happened at the plant

Providing capability to locally operate certain air operated valves (AOVs) following loss of instrument air.

The licensee stated that none of the potential improvements have been incorporated into the plant. The licensee stated that given the low risk as quantified in the IPE, consideration of these improvements has not been a high priority . These improvements may be initiated in conjunction with some related effort, such as accident management.

E.6 Observations

The licensee appears to have analyzed the design and operations of Nine Mile Point 1 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 Nine Mile Point 1; 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 treatment of plant specific initiating events is comprehensive compared with some other IPE/PRA studies. The consideration of leakage from the vessel as affecting the ability to cool the core is notable, especially since Nine Mile Point 1 has emergency condensers which remove decay heat without providing makeup to the vessel. The treatment of recirculation pump seal LOCAs is by far the most comprehensive for any BWR IPE that we have reviewed to date.

One possible shortcoming of the IPE is as follows. The IPE assumes that the core spray pumps, rated for 140 deg. F, can survive up to over 300 deg. F with a 0.5 probability, based on engineering judgement.

Significant level-one IPE findings are as follows:

- station blackout dominates the CDF
- leakage from the vessel impacts CDF
- loss of DHR, defined as the final heat sink, contributes relatively little to CDF.

Station blackout is a major contributor to the CDF; the top two sequences involve station blackout with loss of primary inventory. The ECs do not provide makeup to the primary, and leakage from all sources, including recirculation pump seals, is an important contributor to the CDF. Loss of DHR contributes relatively little to the CDF due to the variety of options available for core cooling during shutdown.

1. INTRODUCTION

1.1 Review Process

This report summarizes the results of our review of the front-end portion of the IPE for Nine Mile Point 1. This review is based on information contained in the IPE submittal along with the licensee's responses to RAI. [IPE] [IPE Responses]

1.2 Plant Characterization

The Nine Mile Point 1 plant is co-located with the Nine Mile 2 plant, but shares few systems with unit 2, because the two plants are different vintage Boiling Water Reactors (BWR). The plant is located on the southeast shore of Lake Ontario in New York, seven miles northeast of Oswego, New York. Nine Mile Point 1 is a BWR 2 with a Mark I containment. General Electric was the Nuclear Steam System Supplier (NSSS); the utility, Niagara Mohawk Power Corporation, was the Architect Engineer (AE) and Stone and Webster was the constructor. The Unit achieved commercial operation in December 1969. The rated power is 1850 MWt and 620 MWe net.

The design features at Nine Mile Point 1 that directly impact the core damage frequency (CDF) are as follows:

- <u>Emergency (isolation) condensers (EC)</u> The ECs require no electrical power to provide for core cooling, and this tends to decrease CDF. However, the ECs alone provide no makeup to the vessel and thus excessive leakage or unisolated seal Loss of Coolant Accidents (LOCAs) cannot be mitigated with the ECs; this tends to increase the CDF.

- <u>Hardened containment vent</u> The hardened containment vent decreases the CDF by providing a backup to loss of containment cooling whereby continued support of core cooling systems can be provided.

- <u>Eight-hour battery lifetime</u> The 8 hour battery lifetime tends to decrease CDF, since it is relatively long compared to battery lifetimes at some other BWRs, thereby allowing for increased likelihood of recovery of offsite power.

- <u>Diesel driven firewater</u> The availability of diesel driven firewater tends to lower the CDF since this source can be used to provide makeup to the ECs and to inject to the vessel via the feedwater piping.

- <u>Ability to power control rod drive (CRD) pumps off 1E power</u> The ability to power the CRD pumps with the diesel generators (DGs) tends to decrease the CDF since this provides an additional source of makeup to the vessel even if offsite power is lost.

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 contains the information requested by Generic Letter 88-20 and NUREG 1335. No obvious omissions were noted. [NUREG 1335][GL 88-20]

The Nine Mile Point 1 IPE is a level 2 PRA. The front-end portion of the IPE is a level I PRA, which is a method that is addressed in Generic Letter 88-20. The specific technique used for the level I PRA was a large event tree/small fault tree technique, and it was clearly described in the submittal.

The submittal described the details of the technique. Internal initiating events and internal flooding were considered. Event trees were developed for all classes of initiating events. The development of component level system fault trees was summarized, and system descriptions were provided. Support systems were modeled with event trees used to establish support state conditions for quantification of the frontline event trees. Inter-system dependencies were discussed in the system descriptions and a table of system dependencies was provided. Data for quantification of the models were provided, including common cause and recovery data. The application of the technique for modeling internal flooding was described in the submittal. No results of uncertainty analyses or sensitivity analyses are provided in the submittal, although the submittal implies that these analyses may have been performed. Importance analyses were performed, and results therefrom are provided in the submittal.

The licensee initiated work on a PRA for Nine Mile Point 1 to fulfill the requests of Generic Letter 88-20. [IPE submittal, Section 1.1]

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

Although Nine Point 1 shares its site with Nine Mile Point 2, there is almost no sharing of systems except for offsite power, and for the ability to use firewater from Unit 2 at Unit 1. A separate IPE was performed for Unit 2.

The submittal states that three types of walkdowns were performed for the IPE. [IPE submittal, Section 1.2] The first walkdown was a week in duration, performed during shutdown, and focused on the primary containment and equipment within primary containment. The second walkdown was performed at power, and it focused on the

reactor building. The third walkdown was a collection of individual walkdowns, performed as necessary in support of modeling of individual systems and in support of the model for internal flooding.

Major documentation used in the IPE included: the UFSAR, procedures, drawings, design basis documents, equipment qualification reports, LERs, in-service test results, and maintenance work reports. [IPE submittal, Section 2.4.1]

The submittal states that many PRAs were reviewed during the preparation of the IPE for Nine Mile Point 1, especially the IPE for Nine Mile Point 2. [IPE submittal, Section 2.4.2] Some of the PRAs reviewed were as follows: [IPE submittal, Section 8]

Brunswick PRA
Shoreham PRA
Limerick PRA
Reactor Safety Study [WASH-1400]
Seabrook PRA.

The IPE did not select a specific freeze date. [IPE Responses] Information was collected as needed throughout the project- from 3/91 through 1/93. The IPE did not credit any pending changes to the plant in the model.

2.1.3 Licensee Participation and Peer Review.

Five full-time engineers from the utility were assigned to the IPE effort. [IPE submittal, Section 5.1] As- needed support was provided from in-line utility organizations, including: operations, nuclear technology, safety analysis, systems engineering, reactor engineering, training, electrical design, fuels, and design basis reconstitution. Consultants from the following organizations were utilized: General Physics, Fauske and Associates, GKA, Haliburton NUS, and EQE.

The submittal states that: "continued application of IPE is a priority for NMPC". [IPE submittal, page 2-3] So far, the IPE has been used by the utility for developing an emergency diesel generator reliability program, supporting operator training, and enhancing operating procedures.

The submittal states that independent in-house reviews were performed throughout the study. [IPE submittal, Section 2.1] As individual IPE packages were prepared, they were reviewed by: Quality Assurance (QA), Independent Safety Engineering Group (ISEG), engineering, operations, maintenance, training, and systems engineering. However, there is no indication that the independent reviews were performed by personnel with PRA expertise to provide an overall review of the IPE. The final IPE package was reviewed by licensing and management. More information on the review efforts is provided in Section 5 2 of the 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 Initiating Events.

The IPE used the following process to identify initiating events: [IPE submittal, Section 2.3]

- review of generic BWR experience
- review of other PRAs
- review of the plant operating history
- failure modes and effects analysis (FMEA) for plant specific initiating events.

The initiating events were quantified as follows. [IPE submittal, Section 3.1.1.1] Anticipated transient initiating events were quantified based on plant specific historical data. Internal (to containment) LOCA events were quantified with generic date from other PRAs. External (to containment) LOCA events were quantified with plant specific analyses. Plant specific support system initiating events were quantified with plant specific fault tree models, except for events involving loss of instrument air and loss of the ultimate heat sink, which were quantified based on plant operating history.

Plant specific initiating events were systematically identified by a FMEA applied to systems. [IPE submittal, Section 3.1.1.2.3] Table 3.1.1-5 of the submittal summarizes the results of the FMEA analysis for the identification of plant specific initiating events. The 23 initiating events analyzed in the IPE, categorized into three initiating event groups, are as follows; [IPE submittal, Table 3.1.1-1]

LOCAs
- Large LOCA in a water line
- Large LOCA in a steam line
- Medium LOCA in a water line
- Medium LOCA in a steam line
- Small LOCA in a water line
- Small LOCA in a steam line
- Interfacing systems LOCAs

Transients
- Reactor scram
- Turbine trip
- Main steam isolation valve (MSIV) closure
- Partial MSIV closure
- Loss of condenser

Total loss of feedwater
Partial loss of feedwater
Inadvertent opening of safety/relief valve

Plant Specific Initiating Events

Loss of offsite power
Loss of operating service water
Loss reactor building closed loop cooling water (RBCLC)
Loss of turbine building closed loop cooling water (TBCLC)
Loss of instrument air
Loss of AC power board 102
Loss of AC power board 103
Loss of lake intake.

Loss of HVAC was not modeled as an initiating event. Based on the analysis of the HVAC systems as summarized in Section 3.2.1.23 of the submittal, the licensee concluded that loss of HVAC has limited impact. This is due to the plant design, specifically: room sizes, equipment sizes, and heat loads. [IPE Responses] The licensee stated that loss of control room cooling does not lead to excessive control room temperatures. [IPE Responses] The IPE model assumed that loss of control room cooling is included in the SCRAM initiating event.

Loss of a DC battery board does not cause plant trip and therefore was not modeled as an initiating event. [IPE Responses]

The IPE includes "loss of lake intake" as an initiating event. This event is loss of the screen house intake water supply, which fails all heat removal capability except that utilizing the isolation condensers . [IPE submittal, Section 3.1.1.2.3] This event has happened once in 23 operating years, and this occurrence was used to quantify the frequency of this initiating event.

The submittal provides a discussion of the identification of interfacing systems LOCAs. [IPE submittal, Section 3.1.1.3.2.2] The licensee concludes that the core spray system is of most potential significance for an interfacing systems LOCA, and assigns an initiating event frequency of 9.2E-6/yr for an interfacing systems LOCA. This frequency does not include the conditional probability the core spray system components exposed to pressures greater than design pressure fail catastrophically. As indicated in Table 3.1.3-1 of the submittal, the probability the core spray components fail catastrophically under exposure to RCS pressure is 1E-4. Therefore, the likelihood of an interfacing systems LOCA that results in significant leakage is small.

Table 3.1.1-1 of the submittal provides point estimate frequencies for the initiating events modeled in the IPE. These frequencies are comparable to values used in typical IPE/PRAs, with two possible exceptions.

The submittal states that recirculation pump seal LOCAs as initiating events, considering the capability to isolate the seal LOCAs, are bounded by the small LOCA initiating event frequency, which is 4E-3/yr for a small LOCA in a water bearing line. [IPE submittal, Section 3.2.1.28-1] [IPE submittal, Table 3.1.1-1] Other PRAs have considered a recirculation pump seal LOCA initiating event as a small-small LOCA and assigned it a higher frequency than that for a small LOCA. For example, the NUREG/CR 4550 study for Peach Bottom used a frequency of 3E-2/yr for the recirculation pump seal LOCA initiating event, and the Browns Ferry IPE used a frequency of 2E-2/yr for the seal LOCA initiating event. [NUREG/CR 4550, Peach Bottom Table 4.3-3] [Browns Ferry IPE submittal, Table 3.1.1-1] The licensee notes that the Nine Mile Point 1 small LOCA model does not address the ability to isolate the LOCA; a seal LOCA can be isolated. The NUREG/CR 4550 analysis for Peach Bottom explicitly considered isolation of the seal LOCA. Therefore, the inclusion of a seal LOCA in the unisolable small LOCA category for Nine Mile Point 1 has minor impact on the overall results. [IPE Responses]

The IPE uses a frequency of 2E-2/yr for spurious opening of a relief valve. This is low in comparison to values used in other IPEs/PRAs for this event. The NUREG/CR 4550 study for Peach Bottom used 0.19/yr, the Browns Ferry IPE used 0.04/yr, the NUREG/CR 4550 study for Grand Gulf used 0.14/yr, and the Cooper IPE used 0.09/yr. [NUREG/CR 4550, Peach Bottom, Table 4.3-1] [Browns Ferry IPE submittal, Table 3.1.1-1] [NUREG/CR 4550, Grand Gulf, Table 4.3-1] [Cooper IPE submittal, Table 3.1.1-14] Furthermore, Nine Mile Point 1 uses electromatic relief valves; other plants using electromatic relief valves have experienced problems with these valves. The Quad Cities plant uses electromatic relief valves; the Quad Cities IPE used a frequency of 0.1/yr for spurious opening of a relief valve. [Quad Cities, IPE Responses]

The frequency assigned to loss of offsite power is 0.05/yr, based on plant data. [IPE submittal, page 3.1.1-8] This plant specific value is lower than a typical generic value. Based on generic data, the frequency of loss of offsite power is estimated to be 0.15/yr. [IPE submittal, page 3.1.1-7]

2.2.2 Event Trees.

Each accident initiating event was included in an appropriate class of initiating events, and each class of initiating events was modeled with an event tree. The following large, frontline event trees were developed: [IPE submittal, Section 3.1.2]

transients and small LOCAs
ATWS
station blackout
medium and large LOCAs.

Two large support system event trees were developed, these being: [IPE submittal, Section 3.1.4]

electrical and actuation support systems event tree
mechanical support systems event tree.

A special event tree was developed for interfacing systems LOCAs. [IPE submittal, Section 3.1.3] The event trees are systemic rather than functional in nature. The mission time used was 24 hours. [IPE submittal, Section 3.1.1.4.1]

The success criteria for no core damage is that the peak core temperature remains below 2500 F. [IPE submittal, Section 3.1.1.4.1] Nine Mile Point 1 does not use jet pumps and the recirculation lines are piped to the vessel such that 2/3 of the core cannot be maintained covered following a large LOCA; thus, collapsed water level is not as meaningful a measure for long term core cooling at Nine Mile Point 1 as it is for BWR plants with jet pumps.

The submittal provides a table of success criteria, with numerous explanatory footnotes, for the various types of accidents . [IPE submittal, Table 3.1.1.8a] These success criteria were developed based on: generic information, plant specific analysis particularly with the MAAP code, and engineering judgement. [IPE submittal, Section 3.1.1.4.2]

It is notable that this IPE modeled seal LOCAs as a result of loss of seal cooling during transient accident sequences. This IPE is only the second BWR IPE that we have reviewed to date that has clearly addressed recirculation pump seal LOCAs due to failure of seal cooling; this is of special interest at BWRs using isolation condensers instead of reactor core isolation cooling (RCIC), such as Nine Mile Point 1.

The success criteria for depressurization with relief valves to allow the use of low pressure cooling systems, specifically core spray or injection with firewater, specify that 2 valves are sufficient if they are opened by operator action when measured level is at the top of the active fuel. [IPE submittal, Table 3.1.1.8a] NUREG/CR-4550 for Peach Bottom required 3 safety relief valves (SRVs) for successful depressurization. [NUREG/CR 4550 Peach Bottom, Table 4.3.4] The licensee discussed the basis for assuming that 2 SRVs can provide successful depressurization. [IPE Responses] MAAP calculations were used to show that 2 SRVs are sufficient. Also, generic calculations in NEDO-24708A indicated that 2 SRVs are adequate.

The licensee also provided information addressing the capability of injection with the firewater system during depressurization with 2 SRVs. [IPE Responses] A calculation indicated that firewater can provide 510 gpm at 130 psid. MAAP calculations

indicated that firewater can be used for core cooling with 2 SRVs providing depressurization.

The success criteria take credit for CRD to provide core cooling. [IPE submittal, Table 3.1.1-8a] The submittal states that one CRD pump at 110 gpm can provide core cooling after 1 hour. [IPE submittal, Page 1.1-2] We performed a calculation to check this statement; our results support the submittal statement. The submittal states that 220 gpm of injection from CRD can provide core cooling immediately after plant shutdown. However, the IPE did not take credit for the use of both CRD pumps to provide 220 gpm of core cooling and thereby mitigate a transient immediately after trip, due to lack of an assessment of the total flow from 2 CRD pumps. [IPE submittal, Table 3.1.1.8a footnote 11] Our calculations indicate that 220 gpm from CRD can provide core cooling immediately after plant trip.

The success criteria state that a large LOCA can be mitigated with 1 core spray pump. [IPE submittal, Table 3.1.1.8a] The table of success criteria does not specify whether or not a core spray topping pump is also required; but, the system description for the core spray system states that both one core spray pump and one core spray topping pump are required. [IPE submittal, page 3.2.1.15-1] The UFSAR states that one core spray pump and one core spray topping pump together can mitigate a large LOCA. [UFSAR, page XV-81a and Tables XV-9A and XV-10]

The success criteria for containment heat removal with containment spray, require 1 containment spray pump, 1 containment spray heat exchanger, and 1 raw water cooling pump for cooling the containment spray heat exchanger. [IPE submittal, Table 3.1.1.8a] This agrees with the UFSAR criteria for containment cooling. [UFSAR, pages XV-169b and XV-169e] If containment cooling with the containment spray system is lost, the success criteria credit the use of shutdown cooling or containment venting for containment heat removal/pressure control. At Nine Mile Point 1, the shutdown cooling system is totally distinct from the containment spray system, utilizing distinct core cooling pumps and heat exchangers, and using service water for a heat sink instead of containment spray raw water.

The IPE credits operator action to vent containment if normal containment heat removal is lost. [IPE submittal, page 3.1.2.1-19] The venting takes place before the drywell pressure limit is reached, at 43.4 psia. [IPE submittal, Section 3.2.1.20.2] If venting is successful, the IPE assumes that core cooling with external sources of water can be continued or initiated, but that recirculation from the suppression pool with the core spray system fails after venting due to loss of adequate net positive suction head available (NPSHA). [IPE submittal, page 3.1.2.4-6]

The submittal states that without normal containment cooling, the procedures call for all injection to containment to be terminated when the drywell pressure limit is reached. [IPE submittal, Section 3.2.1.25.1] This condition is reached if the containment has not been vented. In this situation, only core cooling with

recirculation from the suppression pool with the core spray system is used. The temperature of the suppression pool can reach 312 F before containment fails on overpressure, and the core spray pumps are rated at 140 F. (Section 3.1.1.4.1 of the submittal states that at low temperature, the vent line bellows in the suppression pool wetwell airspace fail at about 65 psig; the saturation pressure at 312 F is about 80 psia which equates to 65 psig.) The IPE assigns a probability of 0.5 for the core spray pumps surviving and continuing to cool the core until containment failure. After containment failure, the core spray pumps are assumed to fail due to loss of adequate NPSHA. [IPE submittal, Section 3.1.2.1] If the core spray pumps fail prior to containment failure, the submittal states that MAAP calculations indicate that containment failure prior to core damage is likely, and the IPE assigns a value of 0.2 for the probability that core damage occurs prior to containment failure. [IPE submittal, Section 3.2.1.25.1] If containment fails before core damage, the IPE credits operator action to cool the core with external sources, in particular: CRD, feedwater, containment spray raw water crosstied to inject via core spray, and firewater crosstied to inject through feedwater. [IPE submittal, Section 3.2.1.25.2] The submittal states the failure of external cooling sources following containment failure due to environmental qualification (EQ) effects is not likely, and a discussion of equipment locations and EQ effects is provided. [IPE submittal, Section 3.2.1.25.2] The IPE assigns a human error failure probability of 0.1 to loss of external injection following containment failure, and states that this human error dominates over EQ related failures.

The submittal provides a discussion of how high suppression pool water temperature can result in failure of the core spray pumps, with loss of normal containment heat removal and no containment venting. [IPE submittal, Section 3.2.1.25.1] Engineering judgement was used to assign a 50% probability that the pumps, with a design temperature of 140 F, can survive up to 312 F. [IPE Responses] As previously discussed, the IPE assumes that even if core spray fails prior to containment failure, there is an 80% probability that containment failure will precede core damage. Thus, with core spray available, the probability that core damage occurs prior to containment failure is 0.5 x 0.2 = 0.1. [IPE submittal, page 3.2.1.25-3] The licensee provided information on the impact of these assumptions (about the core cooling/containment cooling interface) on the CDF. [IPE Responses] If the probability of failure of the core spray pumps without containment cooling is increased from 0.5 to 1.0, the CDF increases by 2%. If core damage is assumed to always occur prior to containment failure, the CDF increases by 16%.

The table of success criteria credits the use of containment spray raw water for core cooling in response to a large LOCA in a water bearing line, and credits the use of either raw water or firewater for core cooling in response to a large LOCA in a steam bearing line. [IPE submittal, Table 3.1.1.8a] These cooling options require operator actions, to either inject containment spray raw water into the core spray lines, or to inject firewater into the feedwater lines. The use of firewater for injection requires installation of a spool piece of piping. [IPE submittal, page 3.2.1.4-2] Firewater is not

credited for the water LOCA since it injects via feedwater into the downcomer and would be lost out the break. It is unlikely that operator actions to implement core cooling with either of these options can be performed quickly enough to prevent core damage following a large LOCA. Blowdown occurs quickly following the large LOCA, and cooling with ECCS must be initiated in a matter of tens of seconds to prevent core damage of uncovered fuel. The large LOCA event tree recognizes this, and does not credit the use of firewater or containment spray raw water unless core spray operates initially. [IPE submittal, page 3.1.2.4-1]

The model for successful mitigation of an ATWS requires that operators inhibit ADS and inject with the liquid poison system. [IPE submittal, Section 3.1.2.2]

The model for station blackout uses a battery lifetime of 8 hours, with load shedding, and considers recovery of offsite power and recovery of failed diesel generators. [IPE submittal, Section 3.1.2.3] Battery power is needed for station blackout events that are accompanied by leakage of coolant from the reactor system such as due to recirculation pump seal failure. For such sequences, battery power is needed to maintain the relief valves open to depressurize the reactor coolant system which, in turn, allows for use of firewater system for low pressure injection makeup. On loss of DC power for such sequences, the relief valves are assumed to reclose, allowing the reactor system to repressurize, rendering low pressure injection ineffective. Operator actions to shed DC loads are required for the batteries to last for 8 hours. If load shedding is successfully completed by 15 minutes, the batteries last for 8 hours; if load shedding is successfully completed after 15 minutes but before 30 minutes, the batteries will last for 4 hours. If DC load shedding is not accomplished by 30 minutes, the batteries fail at 2 hours. If offsite power is not recovered prior to battery depletion, the model assumes that it cannot be recovered and that a DG cannot be started, due to loss of DC control power.

The success criteria for station blackout state that if vessel leakage is less than 25 gpm, then one train of emergency condensers (two condensers per train) with shell makeup (makeup to the secondary sides of the condensers) can provide for core cooling throughout the 24 hour mission time. [IPE submittal, Table 3.1.1.8a and Footnote #8 to Table 3.1.1.8a] This success criteria does not distinguish between shell makeup with the gravity makeup tanks, and long term makeup with condensate or firewater; however, in the discussion of the station blackout event tree, the submittal clarifies the distinction between gravity makeup and long term makeup and discusses the times to core damage with various combinations of makeup failures. [IPE submittal, page 3.1.2.3.6]

The success criteria state that for a stuck open relief valve, if one train of emergency condensers without makeup is available, then depressurization to allow use of low pressure core cooling systems is not required. [IPE submittal, Table 3.1.1.8a] That is, the submittal states that 1 train of emergency condensers without gravity makeup provides core cooling long enough so that when lost, one open relief valve can

maintain pressure sufficiently low to allow use of low pressure core cooling systems, such as: core spray, firewater, and condensate. We compared the energy relieved from one open relief valve at low pressure to the decay heat at the time cooling with the emergency condensers would be lost; based on this comparison, we agree with this success criteria used in the IPE.

2.2.3 Systems Analysis.

System descriptions are included in Section 3.2.1 of the submittal. We found these system descriptions to be informative, and the system schematics to be helpful. Of special note is the system description for the recirculation pump seals.

Our comments on the systems descriptions are as follows.

The system description for AC power discusses the need for operator action to shed loads off the DGs following a DBA, to prevent overloading the DGs. [IPE submittal, Section 3.2.1.2] For example, a containment spray pump needs to be unloaded when containment spray raw water cooling pumps are loaded. [IPE submittal, Section 3.2.1.2.11] The normal supply of offsite power to the 102 and 103 emergency buses is from the station reserve transformers; thus, no transfer of offsite power after plant trip is required.

The system description for firewater states that two pumps are available, one electrical and one diesel. Each pump can provide 2500 gpm at 125 psig; evidently, the pressure is the pump discharge pressure. Firewater from Unit 2 can be crosstied to supply Unit 1. To use firewater for core cooling via injection into the feedwater system, a spool piece of piping must be installed.

The service water system consists of 2 normal pumps and 2 emergency pumps. The normal pumps cool the RBCLC system and the TBCLC system; the emergency pumps only cool the RBCLC system. The emergency service water pumps can be powered off the 1E electrical buses. The source of service water is the screenhouse. There was one event in the history of the plant in which lake water to the screenhouse was lost. As previously discussed in this report, loss of intake water was modeled as an initiating event in the IPE.

The system description for the containment spray system discusses the use of the system for containment cooling by either spray or torus cooling. Operation of the containment spray raw water pumps for cooling the containment spray heat exchangers requires manual operator actions. Operator action can also be used to align two of the four containment spray raw water pumps for injection into the vessel via the core spray system piping.

The RBCLC system cools the following components: recirculation pumps and seals, two instrument air compressors, motor driven feedwater pumps, feedwater booster pumps, and the shutdown cooling system pumps and heat exchangers.

The TBCLC system cools the following components: one instrument air compressor, the service air compressor, and the shaft driven feedwater pump.

The submittal states that the nitrogen system was found to be unimportant and was not modeled. Air is required for opening the outboard MSIVs and for venting containment. The table of system dependencies lists nitrogen as a support system for containment venting presumably as a backup for instrument air. [IPE submittal, Table 3.2.3.2]

The system description for feedwater and condensate discusses the use of the motor driven feedwater pumps as a non-safety grade HPCI system; these pumps require offsite power. One of the motor driven feedwater pumps is operating at power and the other is in standby. A larger feedwater pump, directly driven off the main turbine shaft, is also operating at power.

The system description for the Main Steam System discusses the safety and relief valves, and the MSIVs. The inboard MSIVs are motor operated and fail-as-is; the outboard MSIVs are air operated and fail closed. Six electromatic relief valves on the main steam lines are provided; these valves are distinct from the safety valves, require DC motive power to open, and do not require air to open as do combination safety relief valves used at many BWRs. There are 16 safety valves connected directly to the vessel head.

The system description for the emergency condensers describes the two trains of condensers, each consisting of two condensers, and the gravity-driven secondary makeup tanks, which require operator action to throttle makeup supply. Long term makeup to the secondary shell side of the emergency condensers requires operator action to use the condensate transfer system or the firewater system. At power, the two motor operated isolation valves (one DC and one AC) in each of the two steam lines from the vessel to the emergency condensers are open. At power, each of the two water return lines from the emergency condensers to the reactor vessel are isolated by both a normally closed fail open air operated valve and a check valve.

The system description for the core spray system describes the need for a core spray topping pump to support operation of each of the four core spray pumps. The use of the containment spray raw water crosstie into core spray piping for injection to the vessel is also described.

The system description for CRD describes the makeup flow paths to the vessel, both through the CRD seals and into the 3 inch line connected directly to the vessel. One of the two CRD pumps is normally operating. The majority of the injection is through

the CRD seals. At power, about 44 gpm is injected through the CRD seals, and another 12 gpm is injected into the 3 inch line. After reactor scram, with all control rods fully inserted, about 103 gpm is injected through the CRD seals, and another 9 gpm is injected into the 3 inch line. Thus, after reactor scram, evidently about 110 gpm of CRD is provided to the vessel without the need for operator action to increase CRD flow. With loss of instrument air, air operated valves in the CRD system fail open. [IPE submittal, Section 3.2.1.16-12] The system description and the modeling of CRD in the event trees, imply that operator action is not required to increase CRD flow to the vessel to 110 gpm following reactor trip.

The submittal describes the shutdown cooling system. This system is distinct from the containment spray system. It consists of three trains, each with a shutdown cooling pump and a shutdown cooling heat exchanger. The supplies to the pumps are headered to pull from one recirculation loop suction line, and the returns from the heat exchangers are headered to inject into one recirculation loop discharge line. The system can be used to cool the core when vessel conditions are below 120 psig and 350 F. The shutdown cooling system heat exchangers are cooled by the RBCLC system. The shutdown cooling system pumps can be powered by 1E buses. [IPE submittal, Table 3.2.3.2]

The submittal provides a system description of the containment venting system. Containment can be vented from either the drywell or from the torus airspace. Venting from the torus airspace is preferred since this provides for scrubbing of fission products by the suppression pool water. Procedures direct operators to vent containment before the drywell pressure limit of 43.4 psig is reached. Instrument air is required for operation of the vent. The submittal states that the vent is hardened. [IPE submittal, Section 1.2]

The system description for the liquid poison system states that operator action to inject 30 gpm with one of two pumps is needed to mitigate an ATWS accident scenario. [IPE, Table 3.1.2.2-1] The operator has between 7 minutes and longer than one hour to initiate SLC, depending on the specifics of the accident scenario. The liquid poison system pumps can be powered by 1E buses.

The submittal provides a system description of the recirculation pump seals and cooling for the seals. [IPE submittal, Section 3.2.1.28] If the seals fail as a result of loss of seal cooling, the maximum leakage is expected to be 60 gpm per pump, due to the presence of a breakdown bushing in the seal design. Thus, the maximum expected loss from all five recirculation pumps is 300 gpm. Seal cooling is provided by the RBCLC water system, and is not isolated on containment isolation. [IPE submittal, Pages 3.2.1.22-10 and 3.2.1.22-19] At some BWRs, CRD can also be used to cool the recirc pump seals.

The submittal describes results of tests performed on the seal cartridge. [IPE submittal, Section 3.2.1.28] Based on these tests, the IPE assigns a probability of

0.05 to the probability that a significant seal LOCA develops with loss of cooling to the recirculation pump seals. It is important to note that the potential significance of seal LOCAs is associated with scenarios when core cooling is provided by the emergency condensers with no makeup, and during this state cooling of vessel water using the condensers substantially lowers vessel water pressure and temperature, resulting in less stress on the seals. [IPE submittal, Figure 3.2.1.28-4]

A significant recirculation pump seal LOCA is defined as total leakage greater than 45 gpm, after one hour [IPE submittal, Page 3.1.2.3-4]

The IPE modeled a recirculation pump seal LOCA as a possibility during both general transient events and during station blackout. Following a seal LOCA, makeup to the vessel is required. [IPE Responses]

The submittal includes a discussion HVAC systems, and discusses the need for HVAC to support frontline systems. [IPE submittal, Section 3.2.1-23] Ventilation for the following areas was discussed:

Reactor Building
Battery Rooms and Battery Board Rooms
Diesel Generator Rooms
Emergency Switchgear Rooms
Control Room and Auxiliary Control Room
Screenhouse
Turbine Building.

The submittal states that none of these ventilation systems are required to support operation of equipment. The licensee stated that loss of HVAC is not of item of significance, based on calculations of temperatures resulting from loss of HVAC. [IPE Responses]

The submittal states that the ventilation system for the diesel generator rooms is not needed because the roll doors are normally partially opened during hot weather to provide maximum cooling when the DGs are operating. The doors are closed when the outside temperature drops below 50 F, but that the doors automatically open once the room temperature reaches 80 F and then the doors remain open. [IPE Responses]

2.2.4 <u>System Dependencies</u>.

The submittal provided tables that indicate the dependencies of frontline systems on support systems and support systems on support systems. [IPE submittal, Tables 3.2.3-2 and 3.2.3-1] Also, the submittal provided a table of the dependencies of actuation systems on sensors. [IPE submittal, Table 3.2.3-3]

Important asymmetries in train-level system dependencies were indicated in the submittal, such as differences in sources of cooling water for different air compressors and associated aftercoolers. The following types of dependencies were considered: shared component, instrumentation and control, isolation, motive power, direct equipment cooling, area HVAC (screened out), operator actions, and environmental and phenomenological effects. Our specific comments on the dependency tables follow.

The dependency tables appear complete and are accompanied with numerous informative footnotes.

The system dependency tables indicate that the following pumps are self cooled: core spray, containment spray, CRD, and firewater. The system dependency tables indicate that the following pumps require the indicated external cooling: condensate and feedwater pumps require RBCLC, and feedwater pump 13 also requires TBCLC; and, shutdown cooling system pumps require RBCLC (at the upper temperature limit of operation only).

We have a one minor comment related to the dependency tables. The tables list nitrogen as a support system, yet the system description for nitrogen states that this system was not modeled.

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, if any, were reviewed.

2.3.1 Quantification of Accident Sequence Frequencies.

The Nine Mile Point 1 IPE used the large event tree/small fault tree technique with support states for quantifying core damage. The event trees were systemic. The RISKMAN computer software was used to quantify the fault trees and the event sequences. [IPE submittal, Section 2.3]

Table 3.4.1-2 of the submittal indicates that a truncation value of 1E-11 was used for quantification of all initiating events.

Recovery actions were modeled in the event trees. [IPE submittal, Section 3.2.1.26] The following recovery actions were considered:

- OGR Recovery of offsite power within one hour
- OSP Recovery of offsite power at either 2, 4, or 8 hours conditional on the probability that offsite power was not recovered in the first hour

EDG Recovery of a DG
REC Recovery of a method to remove heat from containment late in an accident.

The event REC considers a number of actions, depending on the preceding failures in the accident sequence, such as: recovery of intake water, recovery of instrument air, and recovery of service water.

2.3.2 Point Estimates and Uncertainty/Sensitivity Analyses.

Mean values were used for point estimate failure frequencies and probabilities. A 24 hour mission time for mitigation of an accident was used.

The submittal implies that an uncertainty analysis was performed using Monte-Carlo techniques; [IPE submittal, Page 2-3] however, no details of an uncertainty analysis are provided in the submittal. The submittal implies that sensitivity analyses were performed; [IPE submittal, page 2-3] however, no details of any sensitivity analyses are provided in the submittal.

Importance analyses were performed and the results of these analyses are provided in the submittal. Event tree top event and split fraction importance measures are tabulated in the submittal [IPE submittal, Tables 3.4.2.1 through 3.4.2.-5] Results are tabulated for: fractional contribution to CDF, risk achievement worth, and risk reduction worth.

2.3.3 Use of Plant-Specific Data.

The entire plant operating history, except for the first year, was used to quantify appropriate initiating events. [IPE submittal, Section 3.1.1.2]. Because of changes to procedures, plant modifications, and equipment aging, only more recent plant specific data were used to quantify failures of mitigating system components. [IPE submittal, Section 3.3.2] To quantify failures of standby components to operate on demand, plant specific data from 1988 through 1992 were used. To quantify failures of operating components, plant maintenance history data from 1984 through 1991 were used. Data from Unit 2 were not used, since the two units are different vintage BWR designs.

Generic data were Bayesian updated with plant specific data, for both hardware failures and maintenance unavailabilities.

The components for which plant specific data were developed are tabulated in Tables 3.3.2.1 and 3.3.2.2 of the submittal.

We performed a spot check of these tables to see if the following components were modeled with plant specific data: diesel generators, ECCS pumps, service water

pumps, motor operated valves (MOVs), AOVs, relief valves, firewater pumps, and circuit breakers. All of these components are listed in the tables except for core spray pumps, containment spray pumps, containment spray raw water pumps, and service water pumps; data for a "general" low pressure centrifugal pump were used for these pumps based on grouping of plant specific data. [IPE Responses]

Table 2-1 of this report compares the plant specific data mean values for selected components from Table 3.3.2-3 of the submittal to values typically used in IPE/PRA studies, using the NUREG/CR 4550 data for Peach Bottom for comparison. [NUREG/CR 4550, Peach Bottom] (The data in the submittal is the Bayesian updated data.)

Table 2-1. Plant Specific Component Failure Data

Component and Failure Mode	Nine Mile Point Value (1), (2) Submittal Table 3.3.2-3	NUREG/ CR 4550 Value (1), (2) Peach Bottom Table 4.9-1
Diesel Generator Fail to Start	2.0E-2/D	3.0E-3/D
Diesel Generator Fail to Run	1.0E-3/H (First Hour) 3.0E-3/H (After First Hour)	2.0E-3/H
Core Spray Pump Fail to Start	1.2E-3/D(3)	3.0E-3/D
Core Spray Pump Fail to Run	4.7E-5/H(3	3.0E-5/H
MOV Fail to Change State (Open/Close)	3.6E-3/D	3.0E-3/D
AOV Fail to Change State (Open/Close)	1.7E-3/D	1.0E-3/D
Relief valve fails to open on demand	8.2E-3/D	1.0E-2(4)
Open valve fails to reclose	3.2E-3/D	9.6E-2/D

(1) D is per demand; these values are probabilities.
(2) H is per hour; these values are frequencies.

(3) Submittal only provides data for centrifugal pumps in general, not for these specific pumps.
(4) Data from NUREG/CR-4550 Generic Data [NUREG/CR-4550, Methodology]

Based on the data in Table 2-1 of this report, the plant specific component failure data are comparable to values used in typical IPE/PRAs.

The IPE calculated plant specific values for recovery of offsite power. [IPE submittal, Section 3.2.1.26.2] We compared this data to that used in a recent PRA. [Surry Shutdown PRA] Figure 2-1 summarizes our comparison. Based on this comparison, the data used for recovery of offsite power in the Nine Mile Point 1 IPE are comparable to that used in typical IPE/PRAs.

2.3.4 Use of Generic Data.

The generic data used for component failures are listed in Table 3.3.1-1 of the submittal. The source of the generic data is primarily the PL&G data base, but the following sources were also used for selected components: the NUCLARR data base of NUREG/CR-4639, the NSAC PRA for Peach Bottom Unit 2, and IEEE STD-500. The generic data are comparable to generic data used in most PRA/IPEs.

The IPE used data from NUREG-1032 to calculate recovery factors for failed diesel generators. [IPE submittal, Section 3.2.1.26.2] The IPE calculated recovery factors for DGs based on a few discrete data points provided in NUREG-1032. Table 2-2 of this report provides the point estimate data from NUREG-1032 as summarized in the submittal.

Table 2-2. DG Recovery Data from NUREG-1032

Time (hours)	Cumulative Probability of Failure to Recover 1 of 2 DGs	Cumulative Probability of Failure to Recover 1 of 1 DGs
0	1.0	1.0
4	0.50	----------
8	----------	0.5
30	<0.10	----------
70	----------	<0.10

We fitted this sparse data using the' Interpolating Polynomial' function in Mathematica, and compared it to the data points used for recovery provided in the table on page 3.2.1.26-4 of the submittal. [Mathematica] Based on this comparison,

the data used in the IPE from a fit of sparse data in NUREG-1032 are comparable to that used in typical IPE/PRAs.

Figure 2-1. Comparison of Data for Recovery of Offsite Power

2.3.5 Common-Cause Quantification.

The method used to model common cause failures was reviewed, and the process by which classes of components were selected for consideration for common cause failures was reviewed.

The submittal states that the method used to model and quantify common cause failure was the Multiple Greek Letter method (MGL), and that the source of data for common cause failures was the PL&G generic common cause database. [IPE submittal, Section 3.3.4]

The submittal states that the identification of systems and components to model for common cause failure was based on NUREG/CR-4780. The IPE considered common cause failures within systems for identical components. Common cause passive failures were neglected.

Table 3.3.4-1 of the submittal tabulates the groups of components by system considered for common cause failure. This list includes such components as circuit breakers, relays, air compressors, and safety/relief valves in addition to the expected components such as diesel generators, pumps, and valves. This list of components selected for consideration is consistent with that used in other IPE/PRA analyses.

Common cause failures across systems were screened from consideration; this is the typical practice used in PRA, except that many BWR IPEs model common cause failure of RCIC and HPCI turbine driven pumps. Nine Mile Point 1 has no RCIC system and HPCI is part of the condensate/feedwater system and does not use steam driven pumps. Therefore, this potential inter-system common cause failure considered for many BWRs is not applicable to Nine Mile Point 1.

Table 3.3.4-2 of the submittal lists the mean common cause failure data MGL factors for the components for which common cause failure was considered. We performed a spot check of this data, as summarized in Table 2-3 of this report.

The common cause values are generally comparable to those used in typical IPE/PRAs, except possibly for the DG fail to start beta factor and the SRV failure beta factor. The licensee provided information about these two cases. The DG combined common cause failure likelihood for failure to start and failure to run results in a model consistent with that used in the Oyster Creek IPE. Also, the CDF for Nine Mile Point 1 is not significantly affected by common cause failure of the DGs. A sensitivity analysis was performed indicating that the CDF is relatively insensitive to common cause failure of the SRVs. [IPE Responses]

Table 2-3. Comparison of Common Cause Failure Factors for 2-of-2 Components

Component	Nine Mile Point 1 Beta Factor Submittal Table 3.3.4-2	Value from Source Indicated in Footnote
Diesel Generator	0.001 (fail to start) 0.01 (fail to run)	0.04 (2), (3) 0.006 (4) (fail to start) 0.03 (4) (fail to run) 0.038(6) (fail to start) 0.004 (7) (fail to start) 0.02 (7) (fail to run)
MOV	0.07	0.05 (1) 0.09 (2), (3) 0.05 (4) 0.09(6) 0.07 (7)
RHR Pump(5)	0.07 (fail to start)(5) 0.01 (fail to run)(5)	0.1 (1), (2) 0.2 (3) 0.1 (4) (fail to start) 0.02 (4) (fail to run) 0.2(6) 0.06 (7) (fail to start) 0.008 (7) (fail to run)
Safety/Relief Valve	0.01(fail to open on demand)	0.1 (1) 0.2 (3) 0.3 (4), (7) (fail to open on pressure) 0.1 (4), (7) (fail to open on signal) 0.2(6) (fail to open)
High Head Pump	------------------------	0.2 (1), (3),(6)
Core Spray Pump(5)	0.07 (fail to start) (5) 0.01 (fail to run) (5)	0.2 (3),(6) 0.2 (4) (fail to start) 0.02 (4) (fail to run) 0.1 (7) (fail to start) 0.009 (7) (fail to run)
Service Water Pump(5)	0.07 (fail to start)(5) 0.01 (fail to run)(5)	0.03 (1), (3),(6) 0.1 (4) (fail to start) 0.02 (4) (fail to run)
Circuit Breaker	0.07	0.2 (4) for 480 V and higher 0.07 (4) for less than 480 V 0.07 (7)

(1) NUREG/CR 4550 Peach Bottom, Table 4.9-1
(2) NUREG/CR 4550 Grand Gulf, Table 4.9-29
(3) NRC IPE Review Guidance, Rev. 1, November 1993
(4) PLG Generic Data in Brown Ferry IPE Submittal Table 3.3.4-10
(5) The Nine Mile Point 1 Submittal only provides factors for a general low pressure centrifugal pump, not for these specific pumps. These factors are the corrected ones provided by the licensee. [IPE Responses]
(6) NUREG/CR 4550 Methodology Document
(7) PLG Generic Data in Fermi IPE submittal Table 3.3-8.

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 success criteria for containment heat removal with containment spray require 1 containment spray pump, 1 containment spray heat exchanger, and 1 raw water cooling pump for cooling the containment spray heat exchanger. [IPE submittal, Table 3.1.1.8a] This agrees with the UFSAR criteria for containment cooling. [UFSAR, pages XV-169b and XV-169e] If containment cooling with the containment spray system is lost, the success criteria credit the use of shutdown cooling or containment venting for containment heat removal/pressure control. At Nine Mile Point 1, the shutdown cooling system is totally distinct from the containment spray system, utilizing distinct core cooling pumps and heat exchangers, and using service water for a heat sink instead of containment spray raw water.

The IPE credits operator action to vent containment if normal containment heat removal is lost. [IPE submittal, page 3.1.2.1-19] The venting takes place before the drywell pressure limit is reached, at 43.4 psia. [IPE submittal, Section 3.2.1.20.2] If venting is successful, the IPE assumes that core cooling with external sources of water can be continued or initiated, but that recirculation from the containment sump with the core spray system fails after venting due to loss of adequate NPSHA. [IPE submittal, page 3.1.2.4-6]

The submittal states that without normal containment cooling, the procedures call for all injection to containment to be terminated when the drywell pressure limit is reached. [IPE submittal, Section 3.2.1.25.1] This condition is reached if the containment has not been vented. In this situation, only core cooling with recirculation from the suppression pool with the core spray system is used. The temperature of the suppression pool can reach 312 F before containment fails on overpressure, and the core spray pumps are rated at 140 F. (Section 3.1.1.4.1 of the submittal states that at low temperature, the vent line bellows in the suppression pool wetwell airspace fail at about 65 psig; the saturation pressure at 312 F is about 80 psia which equates to 65 psig.) The IPE assigns a probability of 0.5 for the core spray pumps surviving and continuing to cool the core up until containment failure. After containment failure, the core spray pumps are assumed to fail due to loss of adequate NPSHA. [IPE submittal, Section 3.1.2.1] If the core spray pumps fail prior to containment failure, the submittal states that MAAP calculations indicate that containment failure prior to core damage is likely, and the IPE assigns a value of 0.2 for the probability that core damage occurs prior to containment failure. [IPE submittal, Section 3.2.1.25.1] If containment fails before core damage, the IPE

credits operator action to cool the core with external sources, in particular: CRD, feedwater, containment spray raw water crosstied to inject via core spray, and firewater crosstied to inject through feedwater. [IPE submittal, Section 3.2.1.25.2] The submittal states the failure of external cooling sources following containment failure due to EQ effects is not likely, and a discussion of equipment locations and EQ effects is provided. [IPE submittal, Section 3.2.1.25.2] The IPE assigns a human error failure probability of 0.1 to loss of external injection following containment failure, and states that this human error dominates over EQ related failures.

The submittal provides a discussion of how high suppression pool water temperature can result in failure of the core spray pumps, with loss of normal containment heat removal and no containment venting. [IPE submittal, Section 3.2.1.25.1] Engineering judgement was used to assign a 50% probability that the pumps, with a design temperature of 140 F, can survive up to 312 F. As previously discussed, the IPE assumes that even if core spray fails prior to containment failure, there is an 80% probability that containment failure will precede core damage. Thus, with core spray available, the probability that core damage occurs prior to containment failure is 0.5 x 0.2 = 0.1. [IPE submittal, page 3.2.1.25-3]

Containment isolation does not result in isolation of cooling to either the recirculation pump motors or to the recirculation pump seals.

Each front end core damage accident sequence was linked directly to the level 2 containment tree, so binning of core damage sequences into Plant Damage States (PDS) was not required. [IPE submittal, Section 3.1.5] However, the submittal does provide a binning classification for core damage sequences to facilitate the assessment of results. These bins are referred to as Core Damage End States, and are tabulated in Table 3.1.5-1 of the submittal.

2.4.2 Human Factors Interfaces.

Based on the front-end review, the following operator actions were noted for possible consideration in the review of the human factors aspects of the IPE:

- action to shed loads off DGs during LOCAs
- action to depressurize during transients with loss of high pressure makeup
- action to inhibit ADS during ATWS
- action to initiate injection with the liquid poison system during an ATWS
- action to provide containment cooling
- action to vent containment
- action to control makeup from gravity tanks to emergency condensers
- action to provide long term makeup to emergency condensers
- action to inject to vessel with firewater via feedwater piping
- action to inject to vessel with containment spray raw water pumps via core spray piping

- action to shed DC loads during station blackout
- recovery of failed DGs.

It appears that all of these actions are in procedures.

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.

Section 3.4.3 of the submittal discusses DHR. The IPE considers DHR to be the final heat sink, and does not include direct loss of core cooling in the definition of loss of DHR. The contribution of loss of DHR to the CDF is 6%. This relatively low contribution is due to the variety of methods available to provide DHR at Nine Mile Point 1, as discussed in Section 2.7.1 of this report.

The licensee performed an importance analysis for loss of DHR. The results of this analysis indicate that operator error is the dominant factor associated with loss of DHR.

No vulnerabilities associated with DHR were found as a result of the IPE.

2.5.2 Diverse Means of DHR.

The IPE evaluated the diverse means for DHR, including:

power conversion system
emergency condensers
containment spray system
shutdown cooling system
containment venting.

The IPE addressed and summarized the means for DHR, using the definition of DHR as the final heat sink.

2.5.3 Unique Features of DHR

Design features at Nine Mile Point 1 that impact the CDF from loss of DHR are as follows:

- Emergency (isolation) condensers (ECs) The ECs require no electrical power to provide for core cooling, and this tends to decrease CDF. However, the

ECs alone provide no makeup to the vessel and thus excessive leakage or unisolated seal Loss of Coolant Accidents (LOCAs) cannot be mitigated with the ECs; this tends to increase the CDF.

- Hardened containment vent The hardened containment vent decreases the CDF by providing a backup to loss of containment cooling whereby continued support of core cooling systems can be provided.

- 8 hour battery lifetime The 8 hour battery lifetime tends to decrease CDF, since it is relatively long compared to battery lifetimes at some other BWRs, thereby allowing for increased likelihood of recovery of offsite power.

- Diesel driven firewater The availability of diesel driven firewater tends to lower the CDF since this source can be used to provide makeup to the ECs and to inject to the vessel via the feedwater piping.

- Ability to power control rod drive (CRD) pumps off 1E power The ability to power the CRD pumps with the diesel generators (DGs) tends to decrease the CDF since this provides an additional source of makeup to the vessel even if offsite power is lost.

2.5.4 Other GSI/USIs Addressed in the Submittal.

The submittal addresses three NRC questions from Generic Letter 91-06 related to resolution of GI A-30, "Adequacy of Safety Related DC Power Supplies". [IPE submittal, Section 3.4.4.1] The licensee requests that GL 91-06 be resolved based on the IPE submittal. [IPE submittal, Section 7.0] Review of the adequacy of the response to these questions is deferred to NRC staff evaluating GI A-30.

No other GSI/USIs are addressed in the submittal.

2.6 Internal Flooding

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

2.6.1 Internal Flooding Methodology.

The following process was used to address internal flooding. [IPE submittal, Section 3.3.8]

Potential flood sources were identified using information on the plant layout combined with a review of industry flooding experience. A plant walkdown was performed to collect additional information, such as the relative locations of equipment in flood zones. Flood propagation was evaluated to determine the extent of equipment

damage from flooding events. Flood scenarios were then conservatively quantified using industry data without credit for operator intervention. For those scenarios surviving this conservative analysis, more detailed analyses were performed, including operator intervention. The general transient event tree was used to evaluate core damage from floods that did not in-and-of-themselves directly cause core damage; this evaluation addressed core damage from a flood initiating event with its resultant failure of equipment, combined with subsequent random failures of equipment.

The model for internal flooding does not consider spray induced failure of equipment. [IPE submittal, Section 3.3.8.3.3] The submittal states that an internal flooding analysis performed by the utility in 1972, and reviewed by the NRC in 1974, concluded that vital equipment is adequately protected from spray and splashing failures by the use of drip-proof motors or shields.

2.6.2 Internal Flooding Results.

The following major flood sources were evaluated: [IPE submittal, Section 3.3.8.3.1]

(1) Systems connected to Lake Ontario
- fire water system
- service water
- circulating water

(2) Condensate Storage Tanks
(3) Suppression Pool
(4) RBCLC System
(5) TBCLC System.

Also, the following flood sources were evaluated:

(6) Containment spray raw water
(7) Diesel generator cooling water
(8) Emergency condenser makeup water tanks
(9) Makeup demineralized water tank.

Sources (6) through (9) were assessed to be insignificant compared to flooding from the circulating and service water system.

Nineteen flooding scenarios are summarized in the submittal. [IPE submittal, Table 3.3.8-1] Of these 19 scenarios, all were screened from further consideration but one, based on a specific evaluation of each flood scenario as summarized in Table 3.3.8-1 of the submittal. Section 3.3.8.3.2 of the submittal provides a description of this one flooding sequence. The initiating event for this sequence is a rupture of the suppression pool torus or the ECCS suction lines, and is assigned a frequency of 1.1E-4/yr. The submittal states that the worst situation resulting from this flood is

flooding of ECCS equipment in all four corner rooms, but that numerous systems remain available for injecting to the vessel, for depressurizing the vessel, and for cooling containment. Based on the number of systems available, this event was screened from quantification for core damage.

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 <u>Dominant Core Damage Sequences</u>.

The IPE utilized systemic event trees, and reported results using the screening criteria from NUREG 1335 for both functional and systemic sequences. [IPE submittal, Section 3.4.1] Sequences were grouped into core damage end states, where the end states are defined based on safety functions lost. Thus, the reporting of the results by end states is a reporting by function. The top 100 core damage systemic sequences from the event trees are also reported in Table 3.4.1-1 of the submittal.

The total CDF from internal initiating events is 5.5E-6/yr. Internal flooding was screened from quantification for core damage. Figure 2-2 summarizes core damage by accident class.

The loss of DHR class of accident refers to sequences in which the final heat sink is lost. The relatively low contribution of loss of DHR to core damage at Nine Mile Point 1 is due to the following attributes. If the power conversion system is unavailable, core cooling can be provided with the emergency condensers which do not require containment cooling for support. Containment can be cooled with any 1 of four containment spray pumps, each with its own heat exchanger. The shutdown cooling system is totally distinct from the containment spray system, even in terms of cooling water for heat exchangers, and it can be used to cool the core without support from any other containment cooling system. Finally, Nine Mile Point 1 has a hardened vent installed which can be used if all other required means of cooling containment are lost, to preserve the ability to cool the core with numerous systems, including: feedwater, CRD, firewater crosstie to feedwater, and containment spray raw water crosstie to core spray. The only negative feature associated with containment cooling at Nine Mile Point 1, is that if the containment is vented or fails by overpressure, cooling of the core using the core spray system in recirculation from the containment sump is lost due to loss of NPSH margin.

Figure 2-2. Core Damage Frequency by Class of Accident

ATWS is a relatively high contributor to core damage at Nine Mile Point 1 compared to some other BWRs. The ATWS model requires inhibition of ADS to mitigate an ATWS; other BWRs have credited level control during an ATWS when ADS is not inhibited and high capacity low pressure ECCS systems actuate.

Station blackout is the dominant class of accident. Nine Mile Point 1 has two DGs, but no station blackout DG or other onsite AC power systems such as combustion gas turbine powered generators. Nine Mile Point 1 has no turbine driven pumps that can inject to the core. The only injection systems that do not require AC power are the emergency condensers and diesel driven firewater, and during station blackout, the emergency condensers require diesel driven firewater for long term secondary makeup. Also, during station blackout a seal LOCA can develop due to loss of seal cooling, and the emergency condensers alone cannot mitigate a seal LOCA sequence since they provide no makeup to the vessel. The battery lifetime is 8 hours with successful load shedding, which is a relatively long lifetime compared to other BWRs.

Loss of injection core damage sequences are dominated by two types of scenarios. In the first type of scenario, high pressure cooling is lost and operators fail to depressurize. In the second type of scenario, a LOCA occurs which requires core spray for core cooling, but core spray fails.

Initiating events which contribute at least 1% to CDF, and their percentage contribution to CDF are as follows:[1]

Loss of Offsite Power leading to Station Blackout	63%
Medium LOCA in Water Line	7%
Loss of Instrument Air	7%
Small LOCA in Water Line	4%
Reactor Scram Initiating Events with ATWS	3%
Turbine Trip with ATWS	2%
Loss of Offsite Power without Station Blackout	2%
Large LOCA in Water Line	2%
Loss of Lake Intake	2%
Loss of RBCLC	1%

The top six core damage sequences are summarized in Table 2-4 of this report.

[1]The reporting of initiating event CDF contributors in the submittal is limited to those initiating events that individually contribute at least 1% to the total CDF. As a group, this set of initiating events represents 93% of the total CDF. The remaining 7% of the CDF appears to be associated with initiating events that individually contribute less than 1% to the total CDF.

Table 2-4. Top 6 Core Damage Sequences

Initiating Event	Dominant Subsequent Failures in Sequence	Percent CDF for Internal CDF For Total Sequence
Loss of Offsite Power	Loss of Both DGs, Failure to Recover Offsite Power in 8 Hours, Failure to Recover either DG in 8 Hours, Inability to Cool Core with ECs alone due to Leakage from Vessel [a], Loss of DC Power to Maintain Relief Valves Open, Loss of Ability to Use Firewater to Inject to Vessel due to Repressurization of Primary when Relief Valves Close	17%
Loss of Offsite Power	Battery Board for Division 1E Power Fails, DG 2 Fails, Offsite Power and DG 2 Not Recovered in 8 hours, Inability to Cool Core with ECs alone due to Leakage from Vessel [a], Loss of DC Power to Maintain Relief Valves Open, Loss of Ability to Use Firewater to Inject to Vessel due to Repressurization of Primary when Relief Valves Close	11%
Loss of Instrument Air	Initiating Event Disables: Power Conversion System, Torus Cooling, and Containment Venting; MSIVs Close Resulting in Isolation of ECs due to Water Entering EC Steam Lines from Feedwater Supply to Vessel; Feedwater Lost due to Operator Error or Equipment failure; CRD Not Considered for Core Cooling unless ECs Operate for 1 Hour; Operators Fail to Depressurize Resulting in Loss of All Injection	5.1%
Medium LOCA in Water Line	Failure of Core Spray Resulting in Inadequate Core Cooling	4.0%
Loss of Offsite Power	Recirculation Pump Seal LOCA Develops thus ECs alone Cannot Cool Core due to Loss of Vessel Inventory; Operators Fail to Load Shed DGs as Required following LOCA, leading to Station Blackout; Offsite Power Not Recovered in 8 Hours; Batteries Fail after 8 Hours; Loss of DC Power to Maintain Relief Valves Open; Loss of Ability to Use Firewater to Inject to Vessel due to Repressurization of Primary when Relief Valves Close	2.4%
Loss of Offsite Power	Relief Valve Sticks Open; Operators Fail to Load Shed DGs as Required, following LOCA leading to Station Blackout; Offsite Power Not Recovered in 8 hours; Batteries Fail after 8 Hours; Loss of DC Power to Maintain Relief Valves Open; Loss of Ability to Use Firewater to Inject to Vessel due to Repressurization of Primary when Relief Valves Close	2.2%

(a) These sequences result in core damage due to leakage from the primary with loss of makeup.

Based on contributions to the CDF, the following systems are most important, listed in decreasing order: [IPE submittal, Section 3.4.2.3]

Emergency AC Power
Emergency DC Power
Relief Valves Reclosing
Diesel Firewater Injection to Vessel.

Based on contributions to CDF, the following operator actions are most important, listed in decreasing order: [IPE submittal, Section 3.4.2.5]

AC power recovery
load shedding DG under LOCA conditions
depressurization
preventing EC isolation and EC recovery after isolation
calibration of core spray injection permissive
feedwater control given loss of instrument air
DC load shedding given station blackout
alignment of torus cooling mode of containment spray.

2.7.2 Vulnerabilities.

Section 3.4.2 of the submittal addresses vulnerabilities. The submittal does not define vulnerability. The submittal concludes: "With regard to core damage frequency, there are no unusual or unique contributors to core damage that suggest a plant specific vulnerability."

2.7.3 Proposed Improvements and Modifications.

Section 6.2 of the submittal discusses improvements judged to have the most potential benefit for reducing risk. These potential improvements are as follows

Load shedding of non-1E battery loads so the non-1E battery system can be used for supply of DC power after the 1E batteries fail at 8 hours during station blackout

Use of a portable battery charger

Improved calibration of low vessel pressure ECCS permissive sensors

Use of plant specific data exclusively for failure of a relief valve to reclose instead of generic data updated with plant specific data, since failure to reclose has never happened at the plant

Providing capability to locally operate certain AOVs following loss of instrument air.

Section 6.3 of the submittal provides additional insights that require more research to validate their cost benefit.

The submittal states that the insights with the most potential for plant improvement may be considered in future plant decision making. [IPE submittal, Section 6.2]

The licensee states that none of the potential improvements have been incorporated into the plant. [IPE Responses] The licensee states that given the low risk as quantified in the IPE, consideration of these improvements has not been a high priority. These improvements may be initiated in conjunction with some related effort, such as accident management.

The licensee states that the IPE model included the prior plant response to meet the station blackout rule. [IPE Responses] No changes, independent of the prior response to meet the station blackout rule, were noted in the IPE as leading to significant risk reduction for station blackout.

3. CONTRACTOR OBSERVATIONS AND CONCLUSIONS

This section of the report provides our overall evaluation of the quality of the front-end portion of the IPE based on this review. Strengths and shortcomings 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 treatment of plant specific initiating events is comprehensive compared with some other IPE/PRA studies. The consideration of leakage from the vessel as affecting the ability to cool the core is notable, especially since Nine Mile Point 1 has emergency condensers which remove decay heat without providing makeup to the vessel. The treatment of recirculation pump seal LOCAs is by far the most comprehensive for any BWR IPE that we have reviewed to date.

One possible shortcoming of the IPE is as follows. The IPE assumes that the core spray pumps, rated for 140 deg. F, can survive up to over 300 deg. F with a 0.5 probability, based on engineering judgement.

Based on our review, the following modeling assumptions have an impact on the overall CDF:

(a) assignment of 0.05 for the probability of a seal LOCA with loss of cooling to the recirculation pump seals
(b) the assumption that no HVAC or ventilation systems are required
(c) the assumption that the core spray pumps, rated for 140 deg. F, can survive up to over 300 deg. F with a 0.5 probability.

All of these assumptions tend to lower the CDF. The first assumption make a seal LOCA relatively unlikely. The second assumption means that all HVAC and ventilation systems can fail and not impact the ability to prevent core damage. The third assumption decreases the likelihood of loss of core cooling given loss of containment heat removal.

Significant level-one IPE findings are as follows:

- station blackout dominates the CDF
- leakage from the vessel impacts CDF
- loss of DHR, defined as the final heat sink, contributes relatively little to CDF.

Station blackout is a major contributor to the CDF; the top two sequences involve station blackout with loss of primary inventory. The ECs do not provide makeup to the primary and leakage from the vessel with no makeup is an important contributor to the CDF. Loss of DHR contributes relatively little to the CDF due to the variety of options available for core cooling during shutdown.

4. DATA SUMMARY SHEETS

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

Overall CDF

The total CDF from internal initiating events is 5.5E-6/year. Internal flooding was screened from quantification for CDF.

Dominant Initiating Events Contributing to CDF

Initiating events that contribute the most to CDF, and their percent contribution, are as follows:

Loss of Offsite Power leading to Station Blackout	63%
Medium LOCA in Water Line	7%
Loss of Instrument Air	7%
Small LOCA in Water Line	4%
Reactor Scram Initiating Events with ATWS	3%
Turbine Trip with ATWS	2%
Loss of Offsite Power without Station Blackout	2%
Large LOCA in Water Line	2%
Loss of Lake Intake	2%
Loss of RBCLC	1%

Dominant Hardware Failures and Operator Errors Contributing to CDF

Based on contributions to the CDF, the following systems are most important, listed in decreasing order:

Emergency AC Power
Emergency DC Power
Relief Valves Reclosing
Diesel Firewater Injection to Vessel.

Based on contributions to CDF, the following operator actions are most important, listed in decreasing order:

AC power recovery
load shedding DG under LOCA conditions
depressurization
preventing EC isolation and EC recovery after isolation
calibration of core spray injection permissive
feedwater control given loss of instrument air
DC load shedding given station blackout

alignment of torus cooling mode of containment spray.

Dominant Accident Classes Contributing to CDF

The submittal summarizes the contribution to overall CDF by accident category, as follows:

Station Blackout	64%
Loss of Injection	19%
ATWS	10%
Loss of DHR	6%.

Design Characteristics Important for CDF

Design features at Nine Mile Point 1 that impact the Core Damage Frequency (CDF) from loss of DHR are as follows:

- Emergency (isolation) condensers (EC) The ECs require no electrical power to provide for core cooling, and this tends to decrease CDF. However, the ECs alone provide no makeup to the vessel and thus excessive leakage or unisolated seal Loss of Coolant Accidents (LOCA) cannot be mitigated with the ECs; this tends to increase the CDF.

- Hardened containment vent The hardened containment vent decreases the CDF by providing a backup to loss of containment cooling whereby continued support of core cooling systems can be provided.

- Eight-hour battery lifetime The 8 hour battery lifetime tends to decrease CDF, since it is relatively long compared to battery lifetimes at some other BWRs, thereby allowing for increased likelihood of recovery of offsite power.

- Diesel driven firewater The availability of diesel driven firewater tends to lower the CDF since this source can be used to provide makeup to the ECs and to inject to the vessel via the feedwater piping.

- Ability to power Control Rod Drive (CRD) pumps off 1E power The ability to power the CRD pumps with the Diesel Generators (DG) tends to decrease the CDF since this provides an additional source of makeup to the vessel even if offsite power is lost.

Modifications

The following potential improvements were noted based on insights from the IPE:

Load shedding of non-1E battery loads so the non-1E battery system can be used for supply of DC power after the 1E batteries fail at 8 hours during station blackout

Use of a portable battery charger

Improved calibration of low vessel pressure ECCS permissive sensors

Use of plant specific data exclusively for failure of a relief valve to reclose instead of generic data updated with plant specific data, since failure to reclose has never happened at the plant

Providing capability to locally operate certain AOVs following loss of instrument air.

The submittal states that the insights with the most potential for plant improvement may be considered in future plant decision making.

Other USI/GSIs Addressed

The submittal addresses three NRC questions from Generic Letter 91-06 related to resolution of GI A-30, "Adequacy of Safety Related DC Power Supplies". [IPE submittal, Section 3.4.4.1] The licensee requests that GL 91-06 be resolved based on the IPE submittal. [IPE submittal, Section 7.0]

Significant PRA Findings

Significant level-one IPE findings are as follows:

- station blackout dominates the CDF
- leakage from the vessel impacts CDF
- loss of DHR, defined as the final heat sink, contributes relatively little to CDF.

Station blackout is a major contributor to the CDF; the top two sequences involve station blackout with loss of primary inventory. The ECs do not provide makeup to the primary and leakage from the vessel with no makeup is an important contributor to the CDF. Loss of DHR contributes relatively little to the CDF due to the variety of options available for core cooling during shutdown.

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[Browns Ferry IPE] IPE Submittal for Browns Ferry

[Fermi IPE] IPE Submittal for Fermi

[Quad Cities IPE] IPE Submittal for Quad Cities

[Cooper IPE] IPE Submittal for Cooper

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