BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2,

INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(FRONT-END)

Enclosure 2

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Brunswick

Technical Evaluation Report on the Individual Plant Examination Front End Analysis

NRC-04-91-066, Task 28

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

E.1 Plant Characterization

Brunswick is a dual unit site located in North Carolina, about 20 miles south of Wilmington. Both units are boiling water reactor (BWR) 4 reactors with Mark I containments. General Electric (GE) provided the nuclear steam supply system (NSSS); United Engineers and Constructors was the architect-engineer (AE) for both units, and Brown and Root was the Constructor. The units achieved commercial operation in 1975 and 1977. Rated power for each unit is 2436 megawatt thermal (MWt) and 821 net megawatt electric (MWe).

Design features at Brunswick that impact the core damage frequency (CDF) are as follows:

- Ability to crosstie 1E buses between units. The ability to crosstie power between units is a beneficial feature which tends to lower CDF in that it allows loss of power at a single unit to be mitigated by providing power from the other unit.

- Ability to vent containment using containment atmospheric control system and standby gas treatment system, but need to stop venting prior to reaching atmospheric pressure to preserve adequate net positive suction head available (NPSHA) for emergency core cooling system (ECCS) pumps pulling from the suppression pool. The ability to vent containment is a beneficial feature that tends to lower CDF, since containment venting is a backup to containment cooling.

- Ability to flood core/containment with service water or diesel driven fire pump via the RHR system. The ability to inject water to the core with service water and diesel driven fire pump is a positive feature that reduces CDF. These systems can be used to cool the core if the low pressure ECCS systems have failed. Fire main water can be provided by a diesel driven pump that can be used during station blackout until DC power is lost and the safety relief valves (SRVs) close.

E.2 Licensee's IPE Process

The Individual Plant Examination (IPE) is an update of a level 1 probabilistic risk assessment (PRA) completed in 1988. The IPE updated the level 1 PRA and performed a back-end analysis that extended the PRA to a level 2 PRA.

The IPE submittal is dated August 1992. Since the date of this submittal, the licensee has updated the IPE model to address several plant modifications and analysis refinements. However, the reporting of results associated with the updated IPE model was limited to a revised total CDF and revised estimate of the station blackout contribution to CDF. Because a complete set of analysis results from the revised IPE model was not available, emphasis was given to reviewing the results and findings from the original submittal. Available results from the updated IPE model are, however, also reported in this review. The freeze date for the original IPE analysis is January 1, 1992.

Utility personnel were involved in about 50% of the effort for the 1988 PRA and were involved in over 50% of the update effort for the IPE. All IPE work was directed by the utility. Contractors used for the IPE update were: NUS, Inc. (NUS), Risk Management Associates (RMA), and SAROS, Inc. (SAROS).

A walkdown for the flooding analysis was performed specifically for the IPE. Results of prior walkdowns for design basis reconstitution were utilized for the internal events analysis.

Major documentation used in the IPE included: the Updated Final Safety Analysis Report (UFSAR), system descriptions, plant procedures, drawings, discussions with operators, maintenance personnel and system engineers, and the Nuclear Regulatory Commission (NRC) review of the earlier PRA.

The original Brunswick PRA had been reviewed by the NRC, and the comments from this review were incorporated into the IPE. Internal reviews of the IPE among plant operations training staff, system engineers, and other utility staff were performed. An external review was conducted by experts in PRAs of BWRs.

The submittal indicates that the licensee intends to maintain a "living" PRA which can be used in the day-to-day resolution of important plant safety issues.

E.3 Front-End Analysis

The methodology chosen for the Brunswick IPE was a Level 2 PRA; for the front-end analysis, the small event tree/large fault tree technique with fault tree linking was used and quantification was performed with the Cut Set and Fault Tree Analysis (CAFTA) software.

The IPE quantified 15 initiating events: 6 loss of coolant accidents (LOCAs), 3 plant specific support system failures, and 6 generic transients. The IPE developed 14 systemic event trees to model the plant response to internal initiating events.

Core damage was defined as peak cladding temperature in excess of 2200 F.

Success criteria were based on past PRAs, and analyses with the MARCHRMA and SAFER/GESTR computer codes.

Support system dependencies were modeled in the fault trees. A table of inter-system dependencies was provided in the submittal.

The IPE used plant specific data to Bayesian update generic data for selected components.

The multiple Greek letter (MGL) method was used to model common cause failures. The data for common cause failures were taken from a combination of standard sources. Common cause failures were modeled within systems.

The technique used to evaluate internal flooding is described in the submittal. Flood scenarios were identified based on the source of flooding, flood propagation, and key equipment locations. Scenarios with little or no impact were screened from further analysis. The remaining scenarios were quantified by combining the flooding effects with independent failures, using the internal initiating event transient event tree modified to account for the specific effects of the flood. Eleven event trees for specific flood initiating events were developed. Both submergence and spray related failures were addressed in the flooding analysis.

As previously noted in Section E.2 of this report, the licensee has updated the IPE model subsequent to the date of the IPE submittal (August 1992) to address several plant modifications and analysis refinements. However, the reporting of results associated with the updated IPE model is incomplete. Therefore, emphasis was given to reviewing the results and findings from the original submittal, though available results from the updated IPE model are also reported. Except where otherwise explicitly noted, the results reported in this review are based on the August 1992 IPE submittal.

The total CDF from internal initiating events and internal flooding is 2.7E-5/yr for each unit, and the total CDF from internal flooding is 1.9E-6/year for each unit.[1]

The top two initiating event contributors to CDF are: loss of offsite power and transients with closure of the main steam isolation valves (MSIVs). Loss of offsite

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[1] The total CDF per unit from the revised IPE model is 1.1E-5/yr.

power with station blackout contributes 66% to total CDF.[2] Loss of decay heat removal (DHR) - defined as loss of the power conversion system (PCS), containment cooling, and containment venting - contributes 25% to overall CDF.[3] Dominant mitigating system failures include:

- failure of the diesel generators (DGs)
- failures in the instrument air system
- failures in the residual heat removal (RHR) system
- failures in the service water system.

Important recovery actions are:

- recovery of the PCS
- recovery of offsite power
- recovery of service water.

Major classes of accidents contributing to the total CDF, and their percent contribution are as follows:

• loss of offsite power followed by station blackout	66%
• transient with loss of DHR	25%
• internal flood	7%
• anticipated transient without scram (ATWS)	3%
• transient with loss of high pressure injection	1%
• LOCA	<1%
• interfacing systems LOCA	<<1%.

Core damage sequences were binned into Plant Damage States (PDSs) for back-end analysis with Containment Event Trees (CETs). The PDSs group core damage sequences based on the conditions in the reactor coolant system (RCS) at the time of core damage, the status of core cooling systems at the time of core damage, and the status of containment systems at the time of core damage. The binning of core damage accident sequences into PDSs is comparable with typical IPE/PRAs.

Based on our review, the following modeling assumption used in the IPE has an impact on the overall CDF:

- no loss of adequate NPSHA for ECCS pumps pulling from the suppression pool if containment venting is controlled

[2] The station blackout CDF contribution is about 44% per the revised IPE model.

[3] Does not include DHR contribution from internal flood. If internal flooding sequences are included, DHR represents approximately 30% of the overall CDF.

The assumption that controlled venting can preserve NPSH margin for using low pressure ECCS pumps pulling from the suppression pool lowers the CDF associated with loss of containment cooling sequences. If controlled venting cannot prevent loss of NPSH margin the total CDF increases by 12%.

E.4 Generic Issues

Although the IPE evaluated all aspects of decay heat removal, the specific evaluation of DHR in the IPE is restricted to the final heat sink options: RHR for containment cooling, use of the PCS, or containment venting.

The IPE identifies loss of DHR as a significant contributor to CDF, contributing to about 25% of the overall CDF. Planned improvements will reduce the contribution of loss of DHR to CDF, by increasing the reliability of the onsite power system and by implementing a simpler containment venting system. No vulnerabilities related to DHR were identified.

The IPE proposes to resolve unresolved safety issue (USI) A-17, "Systems Interactions in Nuclear Power Plants", by virtue of the analysis of internal flooding performed for the IPE. The submittal states that all potential flood sources were considered, and the impact of these floods were evaluated. The flooding analysis identified significant flood induced core damage sequences, but none of these sequences dominate overall CDF. Since the IPE shows no vulnerabilities from internal flooding, the IPE proposes that USI A-17 is resolved.

No other safety issues were specifically addressed for resolution in the Brunswick IPE submittal.

E.5 Vulnerabilities and Plant Improvements

The IPE used the guidelines from Nuclear Management and Resource Council (NUMARC) document 91-04 to search for vulnerabilities. The guidelines were used, "to develop criteria which could be used to determine whether PRA reanalysis, modifications, procedure changes, severe accident management guidance development, or no action was appropriate".

No vulnerabilities were identified.

The IPE submittal estimated the impact of the following plant modifications scheduled after the freeze date:

- installation of a station blackout DG
- installation of remote-manual capability to crosstie 1E buses
- installation of dedicated dc power supply for operating switchyard circuit breakers

- installation of a hardened wetwell vent.

It was estimated that these planned modifications, if implemented, would reduce the CDF from station blackout to less than 1E-5/year.

As previously noted, the licensee has updated the IPE model since the original August 1992 submittal. This analysis update specifically addresses the following:

- increase in frequency of loss of offsite power based on actual event in 1993
- credit for new procedures addressing DC power recovery and station blackout, including load shedding to increase battery lifetime
- installation of logic switches for crosstie of 4160 V buses
- installation of the hardened wetwell vent
- updated plant specific data.

The net result of all these changes is a revised core damage frequency of 1.1E-5/yr. The CDF from station blackout decreased from 1.8E-5/yr to 4.8E-6/yr. Based on this re-evaluation, the licensee at present does not plan to add the fifth station blackout diesel generator and does not plan to add dedicated switchyard batteries.

The CDF from DHR in the updated model is 1.8E-6/year, with credit for the hardened containment vent and with credit for injection with CRD at low pressure.

E.6 Observations

We believe that the licensee analyzed the plant design and operations of Brunswick to discover instances of particular vulnerability to core damage. The licensee has developed an overall appreciation of severe accident behavior, understands the most likely severe accidents at Brunswick, has gained a quantitative understanding of the overall frequency of core damage, and has implemented changes to the plant to help prevent and mitigate severe accidents.

No particular strengths or weaknesses of the IPE were noted.

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

- station blackout is the dominant contributor to the CDF; battery depletion is an important contributor to station blackout, as battery depletion results in the loss of RCIC, HPCI, and also disables the SRVs in the closed position so that fire water injection is prevented
- loss of DHR, defined as the ultimate heat sink, is an important contributor to the CDF; important contributors to DHR CDF include common cause failures of service water valves supplying cooling water to the RHR heat exchangers
- venting must be controlled to prevent loss of core cooling using the suppression pool

1. INTRODUCTION

1.1 Review Process

This report summarizes the results of our review of the front-end portion of the IPE for Brunswick. This review is based on information contained in the IPE submittal dated August 1992 [IPE Submittal] along with the licensee's responses [IPE, Responses] to a request for additional information (RAI).

1.2 Plant Characterization

Brunswick is a dual unit site located in North Carolina, about 20 miles south of Wilmington. Both units are BWR 4 reactors with Mark I containments. GE provided the NSSS; United Engineers and Constructors was the AE for both units, and Brown and Root was the Constructor. The units achieved commercial operation in 1975 and 1977. Rated power for each unit is 2436 MWt and 821 MWe (net).

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- Ability to flood core/containment with service water or diesel driven fire pump via the RHR system. The ability to inject water to the core with service water and diesel driven fire pump is a positive feature that reduces CDF. These systems can be used to cool the core if the low pressure ECCS systems have failed. Fire main water can be provided by a diesel driven pump that can be used during station blackout until DC power is lost and the SRVs close.

2. TECHNICAL REVIEW

2.1 Licensee's IPE Process

We reviewed the process used by the licensee in the IPE with respect to the requests of Generic Letter 88-20. [GL 88-20]

2.1.1 Completeness and Methodology

The Brunswick IPE is a level 2 PRA. The submittal is complete with respect to the type of information requested by Generic Letter 88-20 and NUREG 1335.

We assessed the methodology employed in the front-end portion of the IPE. The IPE is a level 2 PRA. The specific technique used for the front-end portion of the PRA was a small event tree/large fault tree technique, and it was clearly described in the submittal.

The submittal described the details of the technique. Support systems were modeled in the fault trees, and fault tree linking was used to quantify accident sequences. The development of component level system failure fault trees was summarized, and system descriptions were provided. Inter-system dependency tables were provided for both support and frontline systems. Data for quantification of the models were provided, including common cause data. The model for recovery was described in the submittal. Sensitivity and importance analyses were performed. The application of the technique for modeling internal flooding was described in the submittal.

The front-end portion of the IPE is an update to an earlier level 1 PRA that has been reviewed by the NRC. [submittal, Section 1.1] [Prior PRA] [NUREG/CR-5465] This earlier PRA included external events. To meet the requests for the IPE, this level 1 PRA was updated and extended to a level 2 PRA.

The IPE submittal is dated August 1992. Since the date of this submittal, the licensee has further updated the IPE model to address several plant modifications and analysis refinements. However, the reporting of results associated with the updated IPE model was limited to a revised total CDF and revised estimate of the station blackout contribution to CDF. Because a complete set of analysis results from the revised IPE model was not available, emphasis was given to reviewing the results and findings from the original submittal. Available results from the updated IPE model are, however, reported in this review. [IPE, Responses]

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

Brunswick is a two unit site. The two units share the following systems and structures: control room, intake structure, electrical power including diesel generators and 1E buses, and control room heating, ventilation, and air conditioning (HVAC). The IPE does consider the requirements to maintain one unit in hot standby while mitigating an accident at the other unit. [IPE Submittal, Section 3.2.2.1]

The IPE model is based on Unit 2, and the submittal states that there are no significant differences for Unit 1 that necessitate the need for a separate model for Unit 1. [submittal, Section 2.4.2] The submittal states that the major design difference between the two units is that the turbine bypass capability for Unit 1 is 22% while the turbine bypass capability for Unit 2 is 88%. [IPE Submittal, Section 1.2.4] The impact of this difference on the performance of the power conversion system post plant trip is discussed in Section 2.2.2 of this report. Dual unit core damage was not calculated. [IPE, Responses]

Plant walkdowns were performed for the flooding study that addressed potential flooding sources, spray effects, and flood propagation. [IPE Submittal, Section 1.3] The models for internal initiating events used the results of previous walkdowns by the engineering staff that had been performed to reconstitute the design basis of the plant. [IPE Submittal, Section 2.4.1.1.2]

Major documentation used in the IPE included: the UFSAR, system descriptions, plant procedures, drawings, discussions with operators, maintenance personnel and system engineers, and the NRC review of the earlier PRA.

The submittal states that other PRAs were reviewed. [IPE Submittal, Section 2.4.1]

The freeze date for the IPE model used to support the August 1992 submittal was January 1, 1992. [IPE Submittal, Section 2.1]

The submittal indicates that the utility plans to maintain a "living" PRA which can be used in the day-to-day resolution of important plant safety issues. [IPE Submittal, Section 2.4.1.1.1]

2.1.3 Licensee Participation and Peer Review

The original PRA was completed in 1987, and the utility performed about half of the total effort, the other half being performed by a contractor, EI International (EI). The utility converted the EI fault tree software to CAFTA software in 1989. The update of the original PRA for the IPE was accomplished by the utility with assistance from NUS, RMA, and SAROS. Most of the update was performed by the utility. [IPE Submittal, Section 5.0]

Section 5.2 of the submittal summarizes the review process. The following reviews have been performed:

- review of the original PRA by the NRC and NRC contractors

- internal reviews by plant operations training personnel of event trees, success criteria, assumptions, and human actions
- internal review by plant system engineers of detailed fault trees and supporting assumptions
- review by Carolina Power & Light Company (CP&L) staff
- external review of the PRA methodology and assumptions by experts in BWR PRAs.

The IPE team studied comments made on the original PRA models by the NRC contractor review. [NUREG/CR-5465] These review comments were considered in the update for the IPE and that the disposition of each comment is retained as part of the on-site documentation of the IPE. [IPE Submittal, Sections 2.4.1.1.1 and 5.2.1]

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 describes the process used to identify initiating events. [IPE Submittal, Section 3.1.1.1.2] First, a review of Electric Power Research Institute (EPRI) document NP-2230, other PRAs, and Nuclear Power Experience was performed to identify initiating events applicable to BWRs in general. The events from these sources were screened for applicability to Brunswick. Then, Brunswick specific initiating events were identified based on a review of the plant history. Support system failures that could lead to unique initiating events were also reviewed. The submittal contains a table summarizing the results of the analysis of support systems as candidates for special initiating events; this table indicates the impact of support system failures and the rationale for: considering these failures as special initiating events, lumping these failures in with previously identified initiating events, or screening these failures from consideration as initiating events. [IPE Submittal, Table 3.1.2] For example, this table discusses failures of AC power buses and of HVAC systems and the resulting impact on the plant. Table 3.1.9 of the submittal provides the final list of initiating events used in the model along with the frequencies assigned to the initiating events.

Initiating events were quantified as follows. Generic data for general transients were Bayesian updated with plant specific data. Plant-specific initiating events were quantified by considering failures of plant-specific components, utilizing fault tree analysis as necessary. LOCAs were quantified using data from the National Reliability Evaluation Program (NREP).

The IPE quantified 15 initiating events: 6 LOCAs, 3 plant specific support system failures, and 6 generic transients. The following plant-specific initiating events were quantified:

- Loss of 125 V DC battery bus 2A1
- Loss of 125 V DC battery bus 2B2
- Loss of nuclear service water.

The IPE screens out loss of noninterruptible instrument air as an initiating event. Loss of interruptible instrument air causes a loss of condensate flow and that loss of noninterruptible instrument air would affect the scram solenoid valves, and these effects are considered in the generic initiating events loss of feedwater and turbine trip, respectively. A loss of both interruptible and noninterruptible instrument air systems is not likely as the noninterruptible instrument air system is powered from the emergency power buses.

The IPE screens out loss of reactor building closed cooling water (RBCCW) as an initiating event. [IPE Submittal, Table 3.1.2] Loss of RBCCW causes loss of cooling to the control rod drive (CRD) pumps and without RBCCW and CRD the recirc pump seals cannot be cooled; thus, loss of RBCCW is a special transient initiating event that can cause a small LOCA. A complete, unrecoverable loss of RBCCW requires a manual scram; partial loss of RBCCW could require a controlled shutdown. In either case, the contribution of loss of RBCCW to the turbine trip initiating event was judged by the licensee to be negligible. Loss of RBCCW was not considered as a small LOCA initiating event due to loss of seal cooling, because such leaks are small, can be detected, and can be isolated. NUREG/CR 4550 Volume 4 treated seal leaks as small-small LOCAs and the consequences were judged to be negligible. [IPE, Responses]

Table 3.1.1 of the submittal indicates that inadvertent actuation of high pressure coolant injection (HPCI) is included in the initiating event category of a general transient.

The IPE includes a recirculation pump seal LOCA initiating event in the small LOCA category. The submittal provides the size ranges of the LOCAs for both water and steamline breaks. A steamline break outside containment was considered as an initiating event. [IPE Submittal, Section 3.1.1.1.2] Without closure of the MSIVs, this event was modeled as leading to core damage. [IPE Submittal, Section 3.1.2.6] A feedwater line break outside containment was screened from consideration as a special initiating event, because the frequency of a feedwater line break outside containment is low, 3E-4/year, and the likelihood that the MSIVs would fail to close is sufficiently low so that the overall sequence involving a feedwater line break with failure to isolate is low. [IPE, Responses]

Interfacing system LOCAs (ISLOCAs) were evaluated. Table 3.1.3 of the submittal summarizes potential interfacing LOCA pathways. The submittal summarizes the quantification of the dominant interfacing system LOCA pathways; best estimate probabilities were used for the conditional probability of failure of piping/components exposed to above design pressure. The frequency for an ISLOCA is 5.1E-8/yr. This low frequency is due to the use of best estimate criteria for failure of piping/components exposed to beyond design pressure, since this lowers the likelihood of the LOCA given failure of isolation valves by a factor of 100 to 1000. The submittal points out that the reported ISLOCA frequency for the Level 2 analysis is 2.8E-08/yr which is based on a previous data base. The submittal further states that the difference ISLOCA frequencies used in the Level 1 and Level 2 analyses do not affect the results or insights of the IPE and changes will be made as appropriate in the ongoing PRA program. [p. 3.1.28 of submittal]

The frequency for loss of offsite power is 0.074/year. The updated IPE model uses a frequency for loss of offsite power of 0.10/year to reflect an actual plant event that occurred at unit 1 in 1993, and that the fraction of offsite power losses that affect both units has been increased from 0.48 in the submittal to 0.695 in the updated model. [IPE, Responses]

The frequencies of the initiating events used in the IPE are comparable with initiating event frequencies used in typical IPE/PRAs.

2.2.2 Event Trees

Each accident initiating event was included in an appropriate class of initiating events, and each class of initiating events had a corresponding event tree logic model. All functions or systems important to the accident sequences were considered. The interfaces among the events in the event tree logic models and the corresponding mitigating systems were clearly indicated. The event tree logic models properly accounted for: time ordered response, system level dependencies, sequence specific effects on system operability- such as environmental conditions, and high level operator actions as appropriate.

The IPE used systemic event trees to quantify accident sequences. The IPE developed 14 event trees to model the plant response to internal initiating events.

The event tree models are consistent with models used in typical IPE/PRAs. The following discussion addresses specific aspects of the event tree models noted as important during our review.

Core damage was assumed if the peak cladding temperature exceeds 2200 F. [IPE, Responses]

The licensee states that the frequency of a main steam line break outside of containment that is not isolated is low, 1E-8/year. [IPE, Responses] This core damage sequence is negligible in terms of the level 1 core damage quantification, but the sequence was retained for level 2 consideration since it involves bypass of containment.

The transient event tree does not address loss of recirculation pump seal cooling leading to a small LOCA. However, loss of seal cooling leads to small leaks which are well-instrumented and can be easily isolated. [IPE, Responses]

The transient success criteria credit the use of CRD for core cooling immediately after reactor trip. [IPE, Table 3.1.10] The success criteria state that both CRD pumps are required for the first 40 minutes, and then only 1 CRD pump is required; the flow capability of CRD is specified as 250 gpm for 2 pumps and 135 gpm for 1 pump in the table of success criteria. The fault tree model requires 2 CRD pumps and does not allow for reduced flow at any later time even though heat removal requirements decrease as a function of time. [IPE, Responses]

The submittal does not discuss requirements to prevent suppression pool overfill when injecting with CRD. [IPE Submittal, Section 3.1.2.1.4] Emergency response procedures direct the operators to drain the suppression pool to the radwaste system or terminate external injection. [IPE, Responses] The IPE assumed that overfill was inconsequential and it was not modeled. Overfill from low capacity systems such as fire water (about 200 gpm) would take well over 24 hours.

The plant has the ability to inject water to the core using the service water crosstie, with either service water or diesel driven firewater; also, the condensate system can be used to inject to the vessel as a low pressure system. [IPE Submittal, System Descriptions] The success criteria for transients do not include alternate injection with condensate, service water or fire water. [IPE Submittal, Table 3.1.10] But, the description of the transient event tree states that low pressure injection with condensate and firewater are considered. [IPE Submittal, Section 3.1.2.1.2]

The success criteria for a transient indicate that depressurization can be accomplished with any 2 of 7 automatic depressurization system (ADS) valves based on the results of analyses summarized in General Electric Document NEDC-30936P. The document indicates that the use of 2 SRVs can prevent peak cladding temperature from exceeding 2200 F. [IPE Submittal, Table 3.1.10] [IPE, Responses] Other IPEs/PRAs for similar plants have assumed that 3 SRVs are required.

The IPE models a stuck open SRV as a medium LOCA and two stuck open SRVs as a large LOCA. [IPE Submittal, Section 3.1.2.1.2] Other PRAs have modeled a stuck open SRV as a small LOCA. The System Description for ADS in the submittal states that an SRV can relieve 856,000 lbm/hr. [IPE Submittal, Section 3.2.1.4.1] Assuming relief of saturated steam at 1100 psig and moody flow, the effective size of an open SRV is 0.103 sq ft. Since 0.1 sq ft is the upper bound size taken for a small LOCA in a steam line, one stuck open SRV can be considered as a medium LOCA, although it is at the extreme lower end of the medium LOCA size range. [IPE Submittal, Page 3.1.10] Two stuck open SRVs are classified as a large LOCA in the IPE; the equivalent area for two open SRVs is about 0.21 sg ft. Based on the size range for a large LOCA in general given in the IPE, 2 SRVS are a medium LOCA and not a large LOCA. However, MARCHRMA analyses performed by the licensee indicate that for 1 open SRV, the minimum collapsed level is 4 feet above the bottom of the core and the peak cladding temperature is 1550 F; subsequent low pressure injection successfully cools the core. [IPE, Responses] As previously discussed, 2 SRVs can allow core cooling with low pressure injection if all high pressure injection is lost; therefore, two open SRVs can be considered as a large LOCA in terms of success criteria requirements, specifically that no high pressure injection is required. Also, MARCHRMA analyses indicate that for two open SRVs with no high head injection, the minimum collapsed level is 8.5 feet above the bottom of the core and the peak cladding temperature is 700 F.

The success criteria for loss of offsite power credit diesel driven firewater for injection. [IPE Submittal, Table 3.1.12] As indicated in Figure 3.2.11.b and Section 3.2.1.6 of the submittal, use of firewater for injection involves using the service water crosstie for injection via the RHR system by opening valves F073 and F075. These are normally closed motor operated valves. Since the valves fail-as-is, local operator action to open these valves is necessary to use firewater for injection.

The steady state temperatures in the reactor core isolation cooling (RCIC) and HPCI pump rooms were evaluated as part of the Brunswick Station Blackout Coping Study. This study determined that the steady state temperatures in the HPCI and RCIC pump rooms without room cooling would be 152 F and 144 F, respectively, which is below the isolation setpoint of 165 F. The pumps are not expected to overheat until at least 200 F; pump bearings may fail between 200 and 240 F. Plant-specific thermal hydraulic analyses indicate that the suppression pool temperature will not exceed 200 F until 8 hours or later during a station blackout event. The RCIC pump trips on a backpressure of 42 psig; plant-specific thermal hydraulic analyses indicate that 42 psig is not reached until at least 6 hours into an accident. Therefore, it not likely that HPCI or RCIC would fail prior to battery depletion during station blackout. [IPE, Responses]

The loss of offsite power event tree credits crosstie of 4160 V AC power from the opposite unit within 1 hour if the DGs fail. [Figure 3.1.10 and Section 3.1.3.1.2] The interdependence of these two events is not clear, since the normal configuration is evidently for 2 RHR and 1 service water pump at each unit to be powered from the opposite unit. [IPE Submittal, Section 1.2.2.3] If the event for loss of DGs is loss of all DGs at both plants then the 4160 V AC crosstie capability is not available; if the event for loss of DGs is for loss of both DGs at one unit then the need for crosstie is not clear since RHR and SW pumps are already powered from the opposite unit, and these systems can be used for low pressure injection.

Interdependence of the events 'loss of DGs' and '4160 V AC crosstie' is assumed in the loss of offsite power event tree, even though the normal configuration is for selected RHR and service water (SW) pumps to be supplied with power from the opposite unit. Although RHR and service water pumps receive power from the opposite unit, other functions must be provided to the blacked-out unit, including: power to battery chargers, RHR suppression pool cooling system motor operated valves, and drywell cooling.

The loss of offsite power event tree for station blackout assumes that if injection with HPCI or RCIC fails, offsite power must be restored within 30 minutes for successful core cooling, allowing for another 30 minutes to restore systems. [IPE Submittal, Section 3.1.3.1.2]

For station blackout sequences, the IPE model assumed that high pressure systems are lost two hours after accident initiation due to battery depletion. Thermal hydraulic analyses performed by the licensee indicate that core damage can be averted if injection is restored within 5 hours after the accident (3 hours following loss of high pressure injection). However, not credit was taken for the additional 3 hours because plant procedures did not exist at the time the IPE was performed to allow credit for recovery of DC power. (As discussed in Section 2.7 of this report, a procedure to recover DC power has now been implemented at the site so that high pressure systems could be operated in a station blackout beyond the two hour period assumed in the IPE.) [IPE, Responses]

The submittal states that 2000 gpm from a single fire water pump is sufficient to provide core cooling to both units following station blackout. [IPE Submittal, Page 3.1.56]

The NRC sponsored review of the original PRA questioned the impact of a unit trip on the availability of offsite power. [NUREG/CR-5465] We could find no discussion of the effect of plant trip on grid stability in the submittal. The UFSAR states that trip of a unit will not affect the supply of offsite power to that unit. [UFSAR 8.2.1.3] A possible item of significance is dual unit trip affecting the grid, but we could not identify a credible dual unit trip initiating event besides loss of offsite power in the first place. We investigated loss of instrument air as a possible source of dual unit trip, but the UFSAR indicates that although instrument air can be crosstied between the two units, this is not done in practice. [UFSAR Section 3.1A.21]

The success criteria for a large LOCA do not address closure of discharge isolation valves in the recirculation lines to prevent injected low pressure coolant injection (LPCI) water from bypassing the core and flowing out the break. [IPE Submittal, Table

3.1.11] However, this omission from the model was judged to have a very minor impact on the overall CDF.

The success criteria for injection following a large LOCA as used in the IPE are either 1 LPCI pump or 1 core spray (CS) pump. Best estimate analyses performed with the SAFER/GESTR code were used to support the assumption that one LPCI pump can mitigate a large LOCA. [IPE, Responses] These analyses considered leakage past the jet pump slip joints where the jet pump drive line connects to the jet pump nozzles.

The success criteria for LOCAs include use of containment spray with RHR as a method for decay heat removal, but the success criteria for transients do not include use of containment spray for decay heat removal. [IPE Submittal, Tables 3.1.11 and 3.1.10] This is due to the fact that the transient and LOCA event trees only include the suppression pool function and not the containment spray function. [IPE, Responses] Since containment spray shares numerous components with suppression pool cooling, the model assumes that if suppression pool cooling is lost then containment spray is also lost.

The IPE assumes that the following events lead directly to core damage: steam line break outside containment that is not isolated, RHR ISLOCA, and vessel rupture. [IPE Submittal, Section 3.1.2.6]

The IPE credits containment venting as a backup to containment cooling to preserve core cooling using recirculation from the suppression pool in response to all transient and all LOCA accidents; [IPE Submittal, Tables 3.1.10 and 3.1.11] thus, containment venting has an important impact on reducing the overall core damage frequency. The submittal states that containment venting is initiated at about 70 psig and that for containment venting to be successful, the containment cannot be vented all the way to 0 psig due to the possibility of losing adequate NPSHA for the ECCS pumps pulling from the suppression pool. [IPE Submittal, Sections 3.2.1.17 and 3.2.1.17.2.1]

The IPE credits operator action to stop venting prior to depressurizing to atmospheric pressure so as to not cause inadequate NPSHA for the ECCS pumps pulling from the suppression pool. However, the documentation available to us for review did not address two items of potential significance related to this important assumption, namely:

- The identification of specific procedures that instruct the operators to stop venting and the pressure at which venting is stopped.

- With repeated cycles of venting and repressurization, the relative concentration of air to steam in the suppression pool wetwell airspace is reduced, and since air provides the partial pressure to maintain the suppression pool subcooled, over time the loss of air and subsequent buildup of steam will saturate the suppression pool resulting in loss of adequate NPSHA for ECCS pumps regardless of operator control of pressure during venting.

Emergency Operating Procedure (EOP) # EOP-01-SEP-01 directs initiation of venting. However, this procedure does not specifically provide guidance on when to terminate venting. [IPE, Responses] To evaluate the impact of repeated venting on the CDF, follow-up discussions were held with the licensee [Venting Discussions]. In these follow-up discussions, three approaches to evaluate the impact of repeated venting were discussed:

(1) estimate the impact on CDF if venting cannot prevent loss of NPSH margin
(2) calculate the time of the second venting relative to the mission time of 24 hours
(3) consider the updated IPE that credited injection with CRD at low pressure (the updated IPE is discussed in Section 2.7.3 of this report).

The licensee provided an estimate of the increase in total CDF if venting cannot prevent loss of NPSH margin. Under this assumption, the total CDF increases by 12%. This probably represents an over-estimate of the increase on CDF, since it assumes that the NPSH margin is lost at the first venting. Consideration of items (2) and (3) above should reduce the impact on the CDF. The licensee has addressed this item of potential significance and is aware of the significance associated with maintaining adequate NPSH margin with repeated venting.

The discussion of the dominant accident sequences TM-W and TE-W in Section 3.4.1.2 of the submittal implies that if containment cooling is lost and containment venting fails, core cooling using high pressure recirculation from the suppression pool does not fail until after the containment overpressurizes. The best estimate containment failure pressure is about 140 psia. [IPE Submittal, Table 4.4-2 and Figure 4.4.2] For these scenarios, the IPE did not take credit for high pressure ECCS pumps pulling from the suppression pool at the high temperatures in the suppression pool that exist at 140 psia. Rather, high pressure makeup during and after containment overpressurization was restricted in the IPE to the use of CRD and fire water pumps, neither of which are supplied by the suppression pool. [IPE, Responses]

The model for mitigation of an ATWS credits turbine bypass for heat removal. The NRC sponsored review of the original PRA questioned whether or not the smaller turbine bypass capability of Unit 1 (22%) is sufficient to mitigate an ATWS. [NUREG/CR-5465] The submittal does not directly address this topic, but it does indicate that the limiting condition for turbine bypass between the two units was modeled, and implies that unless the recirculation pump trip fails following an ATWS (an unlikely event that was screened out), the turbine bypass capability of unit 1 is sufficient. [IPE Submittal, Page 1.2.5]

The ATWS event tree credits operator action to control level to maintain power within appropriate limits following an ATWS for which the operator fails to inhibit ADS. [IPE Submittal, Figure 3.1.16]

2.2.3 Systems Analysis

System descriptions are included in Section 3.2 of the submittal. Our comments on the system descriptions are as follows.

The discussion of the LPCI mode of RHR implies that loop selection logic is not used at Brunswick. Based on Table 6.3.3.1 and Section 7.3.3.1.4.4 of the UFSAR, it seems that loop selection logic is not used presently at Brunswick.

Testing of the RHR heat exchangers in accordance with generic letter 89-13 was not performed prior to the IPE submittal (August 1992), but recent test results show that the heat exchangers provide a sufficient level of performance.

The system description for electrical power indicates that the normal supply of power during operation is from the unit auxiliary transformers and that following plant trip transfer to the startup transformers must occur to maintain supply of offsite power at shutdown. The system description indicates that the 1E buses can be crosstied between the two units.

The system description for electrical power states that HVAC is not needed to support AC or DC switchgear and batteries, based on analyses performed for the IPE. The DGs require cooling from service water. The system description states that the HVAC system that cools the switchgear is not required, based on room heatup analyses. However, the IPE model does require ventilation in the DG rooms to support operation of the DGs. [IPE, Responses]

The system descriptions for RHR and CS indicate that room cooling is required to support operation of these systems. The system description for HVAC states that RHR room coolers are needed for operation of RHR pumps and HPCI pumps, and that CS room coolers are needed for operation of CS pumps. All of these room coolers are served by the service water system.

The system description for CRD states that CRD requires room cooling and that the CRD pumps require cooling from RBCCW.

The HPCI and CS pumps are self cooled; the RHR pumps use service water for cooling of pump seals, but seal cooling is not required for the RHR pumps to remain functional. [IPE, Responses] Also, a plant-specific analysis shows that the RHR water temperature does not exceed the design temperature of the pump seals during accident modes of operation.

The system description for service water implies that intake structure HVAC is not required for successful operation of the service water pumps.

2.2.4 System Dependencies

Table 3.2.1 of the submittal summarizes dependencies among systems. Our specific comments on the dependency table are as follows.

Footnote A to the table states that the effect of failure of air on the ADS system is negligible due to the presence of nitrogen backup and accumulators on the valves. The SRVs (and MSIVs as well) fail closed on loss of air pressure, and the accumulators will leak past the isolation check valves over the long term leading to inability to maintain the valves open if air supply and nitrogen supply are lost. However, the accumulators in the air supply headers were not credited in the IPE for long term operation of the SRVs and MSIVs because of the possibility of leakage. [IPE, Responses] The IPE did credit operator actions to restore instrument air for long-term operation of valves.

Compressed air is required to operate valves for containment venting. However, the valves can be operated by non-interruptible instrument air system standby air compressors, and these air compressors do not depend on turbine building closed cooling water (TBCCW). The lack of a dependence on TBCCW in turn allows this containment venting air supply to be available following a loss of offsite power. [IPE, Responses]

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 also reviewed.

2.3.1 Quantification of Accident Sequence Frequencies

The Brunswick IPE used the small event tree/large fault tree model for quantifying core damage. Support systems were modeled in the fault trees and fault tree linking was used to account for shared components within systems The event trees are systemic. Fault trees were used to develop component level failures for event tree events. A mission time of 24 hours was used. Quantification of linked event trees was accomplished with the CAFTA software. The analysis used a sequence truncation value of 1E-8 and a cut set truncation value of 1E-9. [IPE Submittal, Section 2.3.7]

2.3.2 Point Estimates and Uncertainty/Sensitivity Analyses

Mean values were used for point estimate failure frequencies and probabilities.

Three Importance calculations were performed, Risk Achievement, Risk Reduction, and Fussell Vesely. [IPE Submittal, Section 1.4.1.4] Each one of these 3 importance measures identified the diesel generators as the most important components. Other important components are in the instrument air, service water, and RHR systems.

Sensitivity analyses related to human actions were performed. [IPE Submittal, Section 1.4.5] Each human error probability less that 1E-2 was increased by a factor of 10, and each human error probability greater than 1E-2 was increased by a factor of 3; the overall CDF increased by about a factor of 5. The component failure data was replaced by the Peach Bottom NUREG 4550/CR data and the overall CDF increased by 3%. The maintenance unavailability data was replaced by Peach Bottom NUREG 4550/CR data and the overall CDF decreased by 35%.

2.3.3 Use of Plant Specific Data

The IPE used plant specific data to bayesian update generic data for selected components. [IPE Submittal, Section 3.3.2] Plant specific data for component failures were taken from plant data covering the time period April 1987 through January 1991 for unit 2.

Plant specific data were used to Bayesian update generic data for the following components:

- DGs
- HPCI pump
- RCIC pump
- Service water pumps
- Standby Liquid Control (SLC) pumps
- CRD pumps
- CS pumps
- Diesel fire main pump.

The IPE used plant specific data for system unavailability for testing and maintenance for selected components, but used generic data for many components. [IPE Submittal, Table 3.3.2-2] Plant specific data were not used for the RHR/LPCI pumps because an analysis of the plant-specific data for the RHR pumps was not available in time to be included in the IPE submittal. [IPE, Responses] Since the date of the submittal, the data analysis has been completed and the updated model uses plant-specific data for RHR pumps. The plant-specific failure probabilities for the RHR pumps are slightly higher than the generic values used in the submittal, but the impact is minimal.

We performed a spot check of the data used in the IPE for component failures. The results of this check are summarized in Table 2-1 of this report.

Table 2-1. IPE Data

Component and Failure Mode	Brunswick Value [1], [2] IPE Submittal Tables 3.3.2-1 and 3.3.1-2 (G) indicates Generic Data (P) indicates Plant Specific Data (Updated)	NUREG/CR 4550 Value [1], [2] Peach Bottom Table 4.9-1
Diesel Generator Fail to Start	8.3E-3/D (P)	3.0E-3/D
Diesel Generator Fail to Run	2.5E-3/H (P)	2.0E-3/H
HPCI Turbine Fail to Start	4.7E-3/D (P)	3E-2/D
HPCI Turbine Fail to Run	8.7E-4/H (P)	5E-3/H
LPCI Pump Fail to Start	2.2E-3/D (G)	3E-3/D
LPCI Pump Fail to Run	2.7E-5/H (G)	3E-5/H
Core Spray Pump Fail to Start	1.9E-3/D (P)	3E-3/D
Core Spray Pump Fail to Run	2.7E-5/H (P)	3E-5/H
Motor operated valve (MOV) Fail to Change State (Open/Close)	3.0E-3/D (G)	3E-3/D
Air operated valve (AOV) Fail to Change State (Open/Close)	2.4E-3/D (G)	1E-3/D
RCIC Pump Fail to Start	5.7E-3/D (P)	3E-2/D
RCIC Pump Fail to Run	6.2E-3/H (P)	5E-3/H

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

Based on the data in Table 2-1 of this report, the plant specific component failure data are comparable to other BWR IPE/PRA studies, except possibly for the values used for HPCI and RCIC fail to start and HPCI fail to run which are about a factor of 5 lower than typical data. However, the failure rates for these components reflect actual plant operating experience as reflected in the update of the generic data.

The IPE used a plant specific evaluation to quantify recovery of offsite power. [IPE Submittal, Table 3.3.3-12] The probabilities for not recovering offsite power that were used in the IPE are as follows:

0.47 within 30 minutes
0.26 within 2 hours
0.054 within 7 hours.

Typical data used for non-recovery of offsite power are given in Figure 2-1 of this report. Based on the data from this figure, the IPE values for non-recovery of offsite power are typical of other IPE/PRA studies.

2.3.4 Use of Generic Data

The generic data used for component failures are listed in Table 3.3.1-2 of the submittal. These data are based on an aggregate of generic data from numerous sources listed in Table 3.3.1.1 of the submittal. We performed a spot check of the generic data and the results are listed in Table 2-1 of this report. Based on this check, the generic data are comparable to values typically used in other PRA/IPEs.

2.3.5 Common Cause Quantification

The MGL method was used to model common cause failures. [IPE Submittal, Section 3.3.4] Numerous sources for common cause failure data were used, such as EPRI NP-3967, the Seabrook PRA, and NUREG/CR-2770. [IPE Submittal, Table 3.3.4.1] Beta factors used in the IPE were determined by averaging the data from the various sources. Gamma factors were taken from EPRI NP-3967.

The following components were considered for common cause failures: [IPE Submittal, Section 3.3.4]

- valves (motor operated valves (MOVs) and safety valves)
- pumps (motor driven)
- diesel generators
- heat exchangers
- batteries.

Like other BWR IPE/PRA studies, the Brunswick IPE did not model common cause failure of the HPCI and RCIC pumps. However, there were two events at Brunswick,

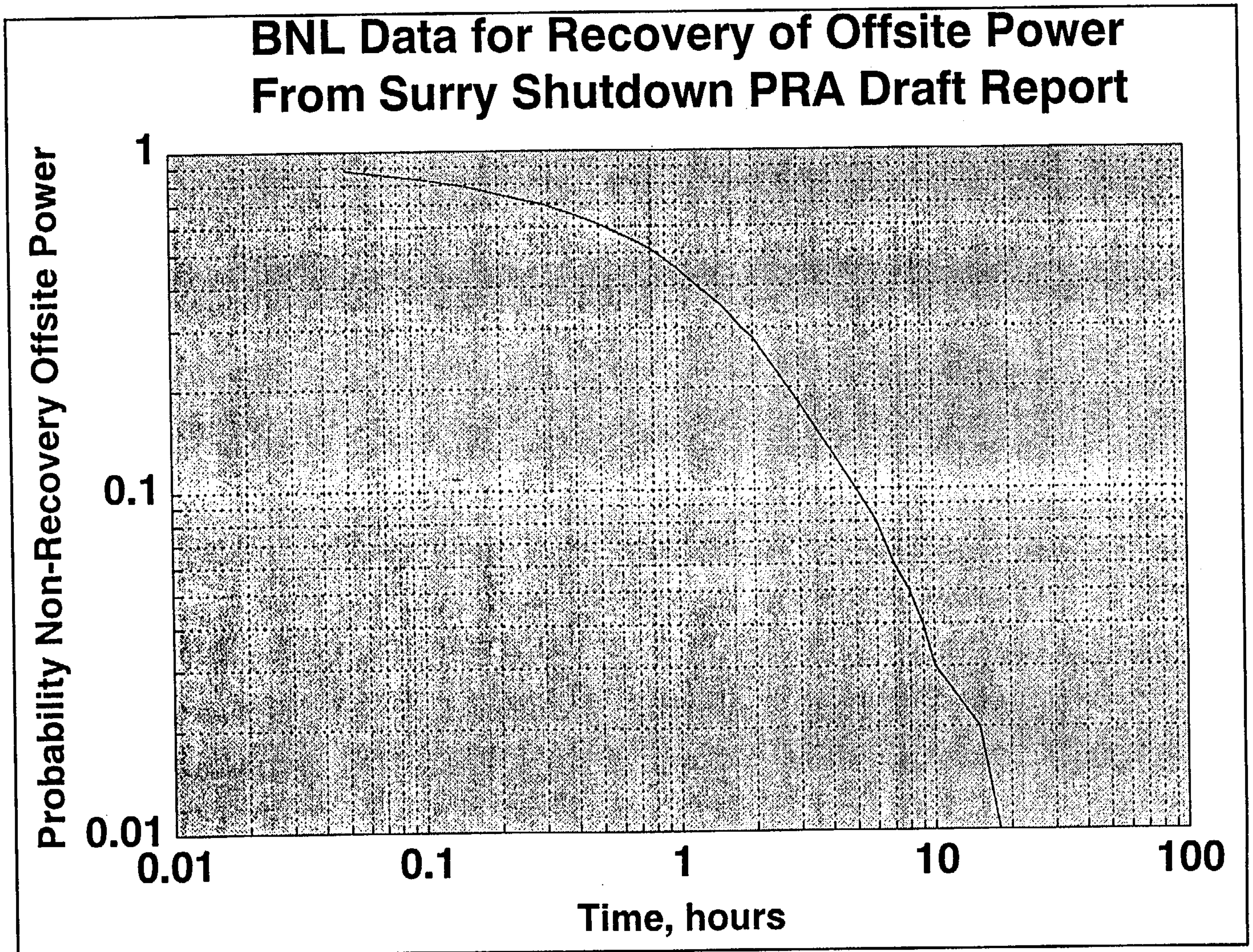

Figure 2-1. Typical Data for Recovery of Offsite Power

in 1980 and 1982, that were attributed to common cause failure of these two systems. The event in 1980 involved mispositioned stop check valves and was due to inadequate procedures and human error. Corrective actions were judged by the licensee to be adequate to prevent recurrence. No similar events have recurred to date. The event in 1982 involved a reference jumper between the speed ramp generator and the governor module. The jumper was missing on the HPCI systems for Units 1 and 2, as well as on the RCIC system for Unit 1. A plant modification was implemented to correct the design problem. The licensee states that common cause failures of HPCI and RCIC were not included in the IPE model because: (1) the corrective measures noted above were judged to prevent future common cause failures, and (2) independent failures of these systems (which are stated to be relatively high from maintenance unavailability and "run" failure rates) would dominant any common cause failures. Table 3.3.2-2 of the submittal indicates that test/maintenance-related unavailabilities of the HPCI and RCIC pumps as used in the IPE model are 0.025 and 0.078, respectively. These test/maintenance unavailabilities are relatively large compared with similar data at other plants. [IPE, Responses]

Table 3.3.4-1 of the submittal lists the common cause failure beta factors used in the model. We performed a spot check of this data, as summarized in Table 2-2 of this report.

Based on the data in Table 2-2 of this report, the common cause factors based on plant specific data appear consistent with other IPE/PRA studies.

2.4 Interface Issues

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

2.4.1 Front-End and Back-End Interfaces

The IPE assumes that if suppression pool cooling is lost, core cooling with ECCS in recirculation from the suppression pool can be maintained by venting of the containment.

The submittal states that containment venting is initiated at about 70 psig and that for containment venting to be successful, the containment cannot be vented all the way to 0 psig due to the possibility of losing adequate NPSHA for the ECCS pumps pulling from the suppression pool. As described more fully in Section 2.2.2 of this report, the licensee provided an estimate of the increase in total CDF if venting cannot prevent loss of NPSH margin. Under this assumption, the total CDF increases by 12%. This probably represents an over-estimate of the increase on CDF, since it assumes that

Table 2-2. Common Cause Factors for 2-of-2 Components

Component	Brunswick Beta Factor IPE Submittal Table 3.3.4-1	Value from Source Indicated in Footnote
Diesel Generator	0.02 (fail to start) 0.03 (fail to run)	0.04 (2), (3) 0.03 (4) fail to run {0.006 for fail to start}
MOV	0.04	0.05 (1) 0.09 (2), (3) 0.05 (4)
RHR Pump	0.09 (fail to start) 0.07 (fail to run)	0.1 (1), (2) 0.2 (3) 0.1 (4) fail to start {0.02 for fail to run}
Safety/Relief Valve	0.1	0.1 (1) 0.2 (3) 0.3 (4) fail to open on pressure {0.1 fail to open on signal}
High Head Pump	0.1 (fail to start, SLC pump) 0.08 (fail to run, SLC pump)	0.2 (1), (3)
Core Spray Pump	0.09 (fail to start) 0.07 (fail to run)	0.2 (3) 0.2 fail to start {0.02 for fail to run}
Service Water Pump	0.06 (fail to start) 0.05 (fail to run)	0.03 (1), (3)
Circuit Breaker	Not provided	0.2 (4) for 480 V and higher 0.07 (4) for less than 480 V

(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 Browns Ferry IPE Table 3.3.4-10.

the NPSH margin is lost at the first venting. [IPE Submittal, Sections 3.2.1.17 and 3.2.1.17.2.1]

The IPE considered seal leakage due to loss of cooling to the recirculation pump seals to be of minor consequence. Spurious containment isolation was not considered to be of significant impact in the IPE model.

Core damage sequences were binned into Plant Damage States (PDSs) for back-end analysis with Containment Event Trees (CETs). [IPE Submittal, Section 1.3] The PDSs group core damage sequences based on the conditions in the RCS at the time of core damage, the status of core cooling systems at the time of core damage, and the status of containment systems at the time of core damage. The PDSs are described in Table 4.3-1 of the submittal. The binning of core damage accident sequences into PDSs is consistent with typical IPE/PRA studies.

2.4.2 Human Factors Interfaces

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

- crosstie of power between the two units
- venting of containment and stopping venting before containment depressurizes
- to the point where adequate NPSHA from the suppression pool is threatened
- initiation of containment cooling with the RHR pumps and heat exchangers
- initiation of SLC following an ATWS
- inhibition of ADS
- manual depressurization
- bypass of high temperature trips for RCIC and HPCI during station blackout where room cooling is lost
- injection with either service water or diesel driven fire main water through the RHR service water crosstie to the RHR system
- injection with both CRD pumps
- actions to restore the main condenser as a heat sink
- actions to stop internal floods
- compensatory actions for loss of HVAC
- recovery of service water.

One unique human action for Brunswick is the need to stop containment venting, once venting is initiated, before containment pressure is reduced to atmospheric, to prevent loss of adequate NPSHA for ECCS pumps pulling from the suppression pool.

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/USI's, if they were addressed in the submittal, were also reviewed.

2.5.1 Examination of DHR

Although the IPE evaluated all aspects of decay heat removal, the evaluation of DHR in Section 3.4.3 of the IPE is restricted to the final heat sink options: RHR for containment cooling, use of the PCS, or containment venting. Thus, the evaluation of DHR does not address direct loss of core cooling.

The submittal discusses items of significance from a DHR analysis of another BWR 4 plant (Cooper), and their applicability to Brunswick. [NUREG/CR-4767] Area 1 deals with station blackout; Brunswick had initially considered addressing this area by adding a fifth DG. Area 2 deals with DC power; Brunswick had also initially considered addressing this area by installing a dedicated battery for the fifth DG, and by providing a battery dedicated to operating switchyard circuit breakers. The third and fourth areas deal with flow diversion from RBCCW and SW; these are not significant at Brunswick.

The IPE identifies loss of DHR as a significant contributor to CDF, contributing to about 25% of the overall CDF. The planned improvements will reduce the contribution of loss of DHR to CDF, by increasing the reliability of the onsite power system and by implementing a simpler containment venting system.

The most important component failures contributing to loss of DHR, listed in decreasing importance are the following: common cause failure to open of SW valves to RHR heat exchangers; RHR loop A unavailable due to maintenance; standby gas treatment valve 2I fails to open for venting; standby gas treatment valve 2N fails to open for venting; RHR heat exchanger A unavailable due to maintenance; and RHR loop B unavailable due to maintenance. [IPE, Responses]

As discussed in Section 2.7.3 of this report, the licensee has re-evaluated the total CDF crediting recent changes to the plant, and the CDF from loss of DHR, as defined in Section 3.4.3 of the submittal, has also been re-evaluated.

2.5.2 Diverse Means of DHR

The IPE evaluated the diverse means for DHR and for core cooling. Cooling options evaluated included: main condenser/feedwater, high and low pressure ECCS systems with containment cooling or containment venting, and core/containment flooding with alternate injection systems such as service water or diesel driven firewater.

The use of containment venting as a backup to suppression pool cooling is an important aspect of DHR modeled in the IPE.

2.5.3 Unique Features of DHR

We reviewed the unique features available for DHR, based on the unique design and operating characteristics of the plant. Unique features of DHR include the use of isolation condensers for cooling at certain BWRs, feed and bleed at PWRs, containment venting at BWRs, and innovative methods to provide cooling to reactor coolant pump or recirculation pump seals.

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

- ability to crosstie 1E buses between units
- ability to vent containment using containment atmospheric control system and standby gas treatment system, but need to stop venting prior to reaching atmospheric pressure to preserve adequate NPSHA for ECCS pumps pulling from the suppression pool
- ability to flood core/containment with service water or diesel driven fire pump via the RHR system.

The impact of these design features on CDF is discussed in Section I.2 of this report.

2.5.4 Other GSI/USI's Addressed in the Submittal

The IPE proposes to resolve USI A-17, "Systems Interactions in Nuclear Power Plants", by virtue of the analysis of internal flooding performed for the IPE. [IPE Submittal, Section 3.4.5.1] The submittal states that all potential flood sources were considered, and the impact of these floods were evaluated. The flooding analysis identified significant flood induced core damage sequences, but none of these sequences dominate overall CDF. Since the IPE shows no vulnerabilities from internal flooding, the submittal proposes that USI A-17 is resolved.

No other safety issues were specifically addressed for resolution in the Brunswick IPE submittal.

2.6 Internal Flooding

We reviewed the process by which the IPE modeled core damage from internal flooding, and we reviewed the results of the internal flooding analysis.

2.6.1 Internal Flooding Methodology

The following process was used to analyze internal flooding: [IPE Submittal, Section 2.3.10.1]

- identification of flood sources
- identification of flood propagation pathways
- determination of important flooding accident sequences
- quantification of core damage from important sequences.

Damage due to spray as well as submergence was addressed in the flooding analysis.

2.6.2 Internal Flooding Results

The transient event tree was modified to reflect system/component unavailabilities due to a flood. Eleven flooding event trees were developed, one for each of 11 flood initiating events

The description in the submittal of the model for internal flooding is included in Section 3.3.8 of the submittal. This description discusses: flood sources, flood propagation, times for operator actions, equipment susceptibility to damage by spray, and scenarios leading to core damage. Table 3.3.8-12 of the submittal summarizes all flood-induced core damage sequences with a frequency greater than 1E-8/year. All of these sequences involve failure of service water lines or gaskets in either the reactor building or the intake structure. The CDF from internal flooding was estimated as 1.9E-6/year, or about 7% of the total CDF.

2.7 Core Damage Sequence Results

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

2.7.1 Dominant Core Damage Sequences

The IPE utilized systemic event trees, but reported results for functional sequences consistent with the criteria delineated in NUREG-1335. [IPE Submittal, Section 3.4.1] The submittal summarizes the contribution to overall CDF from types of accidents initiated by internal events in Table 1.4.1-1 of the submittal. Listed below are the CDF contributors by accident type, including internal flooding.

- loss of offsite power followed by station blackout 66%
- transient with loss of DHR 25%
- internal flood 7%
- anticipated transient without scram (ATWS) 3%
- transient with loss of high pressure injection 1%
- LOCA <1%
- interfacing systems LOCA <<1%.

The CDF from internal initiating events and internal floods is 2.7E-5/year, and the CDF from internal flooding is 1.9E-6/year. [4]

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[4] The total CDF per unit from the revised IPE model is 1.1E-5/yr.

The submittal describes the four functional sequences with a frequency greater than 1E-6/year. [IPE Submittal, Section 3.4.1] These four sequences are summarized in Table 2-3 of this report.

Table 2-3. Functional Sequences with Frequency Greater than 1E-6/year

Initiating Event	Failure of Mitigating Systems	CDF (1/year)
Loss of Offsite Power	Failure of DGs 3 and 4 and failure of power crosstie to Unit 1. Offsite power not restored in 2 hours, HPCI and RCIC fail due to battery depletion. SRVs close after battery depletion rendering all low pressure injection unavailable. Loss of all injection leads to core damage.	1.6E-5
Closure of MSIVs	Failure of containment cooling and failure of containment venting lead to containment failure and loss of core cooling due to inadequate supply of water for ECCS pumps pulling from the suppression pool.	3.3E-6
Loss of Offsite Power	Failure of containment cooling and failure of containment venting lead to containment failure and loss of core cooling due to inadequate supply of water for ECCS pumps pulling from the suppression pool.	2.4E-6
Loss of Offsite Power	Failure of DGs 3 and 4. Random failure of HPCI and RCIC. Failure of crosstie to unit 1 and failure to restore offsite power. Depressurization is successful but diesel driven firewater fails. Loss of all injection leads to core damage.	1.9E-6

Station blackout sequences contribute 66% to the CDF from internal initiating events. [5] Dominant contributors include: failure of DGs, failure of the power crosstie to Unit 1, and battery depletion at 2 hours that renders all high pressure injection unavailable and that renders low pressure injection with the diesel driven firewater pump ineffective due to closure of the SRVs and repressurization of the vessel.

[5] The station blackout CDF contribution is about 44% per the revised IPE model.

Loss of decay heat removal- defined as turbine bypass to the main condenser, containment cooling, or containment venting- contributes about 25% to the total CDF from internal initiating events.

Section 1.4.1.4 of the submittal states that the importance analyses show that the most important components are the DGs, and that components in the following systems are also important: instrument air, service water, and RHR. The submittal does not list important components as determined from the importance analyses, nor does it indicate the reason that failures in the three indicated systems are important.

Failures in these three systems are probably important in that they affect the ability to cool containment and to vent containment.

Section 1.4.3 of the submittal compares the results of the IPE to that of the original PRA for Brunswick, and to the NUREG/CR 4550 PRA for Peach Bottom.

The original PRA for Brunswick calculated a CDF from internal initiating events of 2.1E-5/year, compared to 2.7E-5/year calculated in the IPE which considered both internal initiating events and internal flooding; thus, the two PRAs have similar overall CDFs, but the dominant contributing sequences differ between the two studies.

The original PRA assumed that 5 hours were available to recover offsite power and establish core cooling; the IPE assumes that all the batteries deplete after 2 hours and that without batteries offsite power cannot be easily restored due to loss of DC control power for switchgear CBs. This difference is reflected as an increase in station blackout contribution to CDF from 38% to 66%.

The original PRA used a probability of failure to scram of 3E-5 per demand; the IPE used a value of 4.3E-6. The lower value is stated to be based on a peer review comment from an outside consultant. The original PRA used a probability for operator error to initiate SLC after an ATWS of 3E-2 per demand; the IPE used a value of 2.7E-3 to reflect emphasized operator training related to ATWS. These changes resulted in the contribution of ATWS to CDF decreasing from 44% in the original PRA to 3% in the IPE. [p. 1.4.10 of submittal]

The contribution of loss of DHR increased from 4% in the original PRA to 25%[6] in the IPE, due to more detailed modeling of DHR and incorporation of flooding events in the IPE.

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[6] Does not include DHR contribution from internal flood. If internal flooding sequences are included, DHR represents approximately 30% of the overall CDF.

2.7.2 Vulnerabilities

Section 3.4.2 of the submittal discusses vulnerability screening. The submittal states that the NUMARC Severe Accident Closure Guidelines (NUMARC 91-04) were used, "to develop criteria which could be used to determine whether PRA reanalysis, modifications, procedure changes, severe accident management guidance development, or no action was appropriate". [IPE Submittal, Section 3.4.2.1]

Other criteria were also used to evaluate the IPE results, including cost-effectiveness and impact on plant operations of potential fixes, as viewed by experienced corporate and plant engineers and operations staff. [IPE, Responses]

The submittal does not specifically identify any vulnerabilities.

2.7.3 Proposed Improvements and Modifications

The licensee performed a review of both the important issues identified by the IPE and the list of pending modifications to the plant. Section 1.4.4 of the August 1992 submittal summarizes options based this review that were considered for reducing CDF; these options focus on either reducing the likelihood of station blackout or reducing the likelihood of loss of DHR, since these two classes of accidents contribute 96% to the overall CDF. The options initially considered for reducing the likelihood of station blackout were:

- increase the reliability of the onsite power generation system
- provide a dc power source for operating circuit breakers in the switchyard that is separate from the station batteries
- increase the battery lifetimes by implementing enhanced load shedding.

The options initially considered for reducing the likelihood of loss of DHR were:

- increase the reliability of containment venting
- ensure long term injection from the CST including refill of the CST so that injection with water from outside the suppression pool can be maintained if containment cooling is lost.

Pending modifications initially considered for reducing CDF in the August 1992 submittal were: [IPE Submittal, Section 3.4.2.1 and 1.4.4]

- installation of a station blackout DG
- installation of remote-manual capability to crosstie 1E buses
- installation of dedicated DC power supply for operating switchyard circuit breakers
- installation of a hardened wetwell vent.

The IPE estimated that these modifications would reduce the CDF from station blackout to less than 1E-5/yr, based on sensitivity studies. The addition of a fifth diesel generator alone was estimated to reduce the CDF by about 40%. [p. 1.4-14, Section 6.1.7.1 of submittal]

Since the date of the August 1992 submittal, the licensee has decided to not install the fifth DG and has decided to not install dedicated DC power for operating switchyard breakers. [IPE, Responses] [IPE, Update]

The updated IPE model developed subsequent to the August 1992 submittal specifically addressed the following:

- increase in frequency of loss of offsite power based on actual event in 1993
- credit for new procedures addressing DC power recovery and station blackout, including load shedding to increase battery lifetime
- installation of logic switches for cross tie of 4160 V buses
- installation of the hardened wetwell vent
- updated plant specific data.

The net result of all these changes is a revised core damage frequency of 1.1E-5/yr. The CDF from station blackout decreased from 1.8E-5/yr to 4.8E-6/yr. Based on this re-evaluation, the licensee at present does not plan to add the fifth station blackout DG and does not plan to add dedicated switchyard batteries.

The updated IPE model also credits installation of the hardened wetwell vent and use of CRD injection at low pressure. [IPE, Update] Based on these changes to the model, the licensee states that the CDF from loss of DHR contributes 1.8E-6/year to the revised total CDF of 1.1E-5/year. [IPE, Update]

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 weaknesses of the IPE are summarized. Important assumptions of the model are summarized. Major insights from the IPE are presented.

No particular strengths or weaknesses of the IPE were noted.

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

- no loss of adequate NPSHA for ECCS pumps pulling from the suppression pool if containment venting is controlled
- no common cause failure between the HPCI and the RCIC pump.

Both of these assumptions tend to decrease the overall CDF. The assumption that controlled venting can preserve NPSH margin for using low pressure ECCS pumps pulling from the suppression pool lowers the CDF associated with loss of containment cooling sequences. As discussed in Section 2.2.2 of this report, if controlled venting cannot prevent loss of NPSH margin the total CDF increases by 12%. The assumption that the HPCI and RCIC pumps do not have a common cause failure tends to lower the CDF from transients or small LOCAs.

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

- station blackout is the dominant contributor to the CDF; battery depletion is an important contributor to station blackout, as battery depletion results in the loss of RCIC, HPCI, and also disables the SRVs in the closed position so that fire water injection is prevented
- loss of DHR, defined as the ultimate heat sink, is an important contributor to the CDF; important contributors to DHR CDF include common cause failures of service water valves supplying cooling water to the RHR heat exchangers
- venting must be controlled to prevent loss of core cooling using the suppression pool

4. DATA SUMMARY SHEETS

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

Overall CDF

The CDF from internal initiating events and internal floods is 2.7E-5/year, and the CDF from internal flooding is 1.9E-6/year.[7]

Dominant Initiating Events Contributing to CDF

The top two initiating event contributors to CDF are: loss of offsite power and transients with closure of the MSIVs.

Dominant Hardware Failures and Operator Errors Contributing to CDF

Dominant mitigating system failures include:

failure of the DGs
failures in the instrument air system
failures in the RHR system
failures in the service water system.

Important recovery actions are:
recovery of the PCS
recovery of offsite power
recovery of service water.

Dominant Accident Classes Contributing to CDF

The CDF by class of accident is as follows:

loss of offsite power followed by station blackout	66% [8]
transient with loss of DHR	25% [9]
internal flood	7%
ATWS	3%
transient with loss of high pressure injection	1%
LOCA	<1%
interfacing systems LOCA	<<1%.

[7] The total CDF per unit from the revised IPE model is 1.1E-5/yr.

[8] The station blackout CDF contribution is about 44% per the revised IPE model.

[9] Does not include DHR contribution from internal flood. If internal flooding sequences are included, DHR represents approximately 30% of the overall CDF.

Design Characteristics Important for CDF

Design features at Brunswick that impact the core damage frequency (CDF) are as follows:

- ability to crosstie 1E buses between units
- ability to vent containment using containment atmospheric control system and standby gas treatment system, but need to stop venting prior to reaching atmospheric pressure to preserve adequate NPSHA for ECCS pumps pulling from the suppression pool
- ability to flood core/containment with service water or diesel driven firewater via the RHR system.

Modifications

The licensee performed a review of both the important issues identified by the IPE and the list of pending modifications to the plant. Section 1.4.4 of the submittal summarizes options based this review that were considered for reducing CDF; these options focus on either reducing the likelihood of station blackout or reducing the likelihood of loss of DHR, since these two classes of accidents contribute 96% to the overall CDF. The options initially considered for reducing the likelihood of station blackout were:

- increase the reliability of the onsite power generation system

- provide a dc power source for operating circuit breakers in the switchyard that is separate from the station batteries

- increase the battery lifetimes by implementing enhanced load shedding.

The options initially considered for reducing the likelihood of loss of DHR were:

- increase the reliability of containment venting

- ensure long term injection from the CST including refill of the CST so that injection with water from outside the suppression pool can be maintained if containment cooling is lost.

Pending modifications initially considered for reducing CDF were:

- installation of a station blackout DG

- installation of remote-manual capability to crosstie 1E buses

- installation of dedicated DC power supply for operating switchyard circuit breakers

- installation of a hardened wetwell vent.

The IPE estimated that these modifications would reduce the CDF from station blackout to less than 1E-5/yr, based on sensitivity studies. The addition of a fifth diesel generator alone was estimated to reduce the CDF by about 40%.

Since the date of the submittal (August 1992), the licensee has decided to not install the fifth DG and has decided to not install dedicated DC power for operating switchyard breakers.

The licensee updated the IPE model, specifically addressing the following:

(1) increase in frequency of loss of offsite power based on actual event in 1993
(2) credit for new procedures addressing DC power recovery and station blackout, including load shedding to increase battery lifetime
(3) installation of logic switches for cross tie of 4160 V buses
(4) installation of the hardened wetwell vent
(5) updated plant specific data.

The net result of all these changes is a revised core damage frequency of 1.1E-5/yr. The CDF from station blackout decreased from 1.8E-5/yr to 4.8E-6/yr. Based on this re-evaluation, the licensee at present does not plan to add the fifth station blackout DG and does not plan to add dedicated switchyard batteries.

The updated model also credits installation of the hardened wetwell vent and use of CRD injection at low pressure. Based on these changes to the model, the licensee states that the CDF from loss of DHR contributes 1.8E-6/year to the revised total CDF of 1.1E-5/year.

<u>Other USI/GSIs Addressed</u>

The IPE proposes to resolve USI A-17, "Systems Interactions in Nuclear Power Plants", by virtue of the analysis of internal flooding performed for the IPE. The submittal states that all potential flood sources were considered, and the impact of these floods were evaluated. The flooding analysis identified significant flood induced core damage sequences, but none of these sequences dominate overall CDF. Since the IPE shows no vulnerabilities from internal flooding, the submittal proposes that USI A-17 is resolved.

No other safety issues were specifically addressed for resolution in the Brunswick IPE submittal.

Significant PRA Findings

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

- station blackout is the dominant contributor to the CDF; battery depletion is an important contributor to station blackout, as battery depletion results in the loss of RCIC, HPCI, and also disables the SRVs in the closed position so that fire water injection is prevented
- loss of DHR, defined as the ultimate heat sink, is an important contributor to the CDF; important contributors to DHR CDF include common cause failures of service water valves supplying cooling water to the RHR heat exchangers
- venting must be controlled to prevent loss of core cooling using the suppression pool.

REFERENCES

[GL 88-20] "Individual Plant Examination For Severe Accident Vulnerabilities - 10 CFR 50.54 (f)", Generic Letter 88.20, U.S. Nuclear Regulatory Commission, November 23, 1988.

[NUREG-1335] "Individual Plant Examination submittal Guidance", NUREG-1335, U. S. Nuclear Regulatory Commission, August, 1989.

[IPE Submittal] Brunswick IPE submittal, August, 1992

[IPE, Responses] Letter from R.P. Lopriore, CP&L, to NRC, February 27, 1995, Serial BSEP 95-0104

[Venting Discussions] Two Telcons between NRC, CP&L, and SEA regarding Repeated Venting and Impact on NPSH Margin, April 20 and 21, 1995

[IPE, Update] Letter from R.P. Lopriore, CP&L, to NRC, September 9, 1994, Serial BSEP 94-0359

[Surry, Shutdown] "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1", NUREG/CR-6144, June, 1994.

[Prior PRA] Original BSEP PRA, 1988

[NUREG/CR-5465] "Review of the Brunswick Steam Electric Plant Probabilistic Risk Assessment", NUREG/CR-5465, November, 1989

[UFSAR] Updated Final Safety Analysis Report for Brunswick

[NUREG/CR 4550, Grand Gulf] NUREG/CR- 4550, Vol 6, Rev 1, Part 1, Analysis of Core Damage Frequency: Grand Gulf, Unit 1 Internal Events

[NUREG/CR 4550, Peach Bottom] NUREG/CR- 4550, Vol 4, Rev 1, Part 1, Analysis of Core Damage Frequency: Peach Bottom, Unit 2 Internal Events

[NUREG/CR-4767] "Shutdown Decay Heat Removal Analysis of a General Electric BWR-4 / Mark I"

[TER, Cooper] SEA TER for Cooper IPE