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UNITED STATES
NUCLEAR REGULATORY COMMISSION
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October 8, 1992

MEMORANDUM FOR: Elinor G. Adensam, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulations

FROM: Ronnie H. Lo, Senior Project Manager
Project Directorate II-1
Division of Reactor Projects I/II

SUBJECT: SUMMARY OF BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2, INDIVIDUAL PLANT EXAMINATION

Carolina Power & Light Company (CP&L) submitted the Individual Plant Examination (IPE) report for Brunswick Steam Electric Plant, Units 1 and 2 (BSEP), on August 31, 1992, in accordance with Generic Letter (GL) 88-20. The IPE for BSEP was completed using a plant-specific probabilistic risk assessment (PRA) consistent with the method in Section 4 of GL 88-20. The IPE for BSEP consists of a Level 1 PRA which was analyzed with small event tree/large fault tree methodology, and a Level 2 PRA which was based on a BSEP-specific containment event tree. The original BSEP PRA, submitted to the NRC in April 1988, served as a foundation for the current PRA and the BSEP IPE.

The results of the BSEP Level 1 IPE indicate an overall core damage frequency (CDF) of 2.7E-5/year. Approximately 66% and 30% of the CDF are attributed to station blackout (SBO) (1.8E-5/year) and loss of decay heat removal (LDHR) (8.3E-6/year) sequences, respectively. CP&L has scheduled General Design Criterion 17 (GDC 17) related modifications, including the installation of a 5th diesel generator at the BSEP site that should substantially reduce the contributions to the CDF from the SBO scenarios. In addition, CP&L has plans to install a hardened wetwell vent which should further reduce the CDF contribution associated with the LDHR scenarios.

CP&L has conducted a study on the effects of performance at BSEP on the PRA. CP&L states that the NRC concern which resulted in BSEP being placed on the "Watch List" were primarily related to the lack of administrative control of plant configuration and plant material condition. Studies were performed to determine how these "regulatory concerns" might affect the PRA model; including a sensitivity study on human error probabilities and a sensitivity study on component and maintenance unavailabilities on the CDF. CP&L

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concludes from theses studies that the BSEP IPE is conservative and uncertainties introduced by those concerns "would be within the range of the current (IPE) results".

A copy of the Executive Summary of the IPE is enclosed for your information.

ORIGINAL SIGNED BY:

Ronnie H. Lo, Senior Project Manager
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Enclosure:
Executive Summary

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1.0 EXECUTIVE SUMMARY

1.1 Background and Objectives

In 1986, the Carolina Power and Light Company (CP&L) initiated a probabilistic risk assessment (PRA) for the Brunswick Steam Electric Plant (BSEP) Units 1 and 2, GE BWR-4/Mk I boiling water reactors (BWR) [1-1]. The completed BSEP Level 1 risk analysis, with external events, was submitted for NRC review in 1988 [1-2]. This risk analysis was a natural extension of several previously performed PRA-related activities which included performance of a decay heat removal probabilistic safety study, published in 1985 as NSAC-83 [1-3] and the development of an event tree level plant risk model termed BSEP MAC [1-4].

The BSEP PRA was performed by the On-Site Nuclear Safety Unit of the CP&L Corporate Nuclear Safety Section, with technical support provided by EI International, EQE Incorporated and United Engineers and Constructors.

The objectives for the original PRA were threefold:

(1) To evaluate plant risk, severe accident behavior, and vulnerabilities to severe accidents for BSEP Units 1 and 2.

(2) To transfer in-depth PRA technology to CP&L personnel so that they would be able to keep the PRA up to date.

(3) To develop detailed microcomputer-based plant risk models which could be routinely used by CP&L technical staff for quantitative and qualitative insights which would assist in the resolution of licensing and day-to-day operational issues.

CP&L has continued to commit manpower and financial resources towards the maintenance, modification and re-evaluation of the PRA models to reflect changes in the plant design and operational practices which have been implemented since the completion of the PRA in 1988. The PRA program has been successful in developing internal PRA and severe accident expertise, in identifying and improving deficiencies in severe accident prevention and mitigation, and, in supporting the risk management requirements of day-to-day operations.

In 1988, the US NRC issued Generic Letter No. 88-20 which directed nuclear plant licensees to perform an Individual Plant Examination (IPE) for each of their power plants. The letter indicated that strong utility involvement in the development of the PRA was of paramount importance so that the utility staff would be able to:

(1) develop an overall appreciation of severe accident behavior,

(2) understand the most likely severe accident sequences that could occur;

(3) gain a more quantitative understanding of the overall probability of core damage and radioactive material releases; and,

(4) if necessary, reduce the overall probability of core damage and radioactive material release by appropriate modifications to procedures and hardware that would help prevent or mitigate severe accidents.

Based on the success and in-house acceptance of the results from the original CP&L risk program, knowledge of the IPE requirements and an understanding that risk based resolution of Mark I containment issues would be needed, CP&L management decided to perform a full Level 2 risk analysis for the Brunswick IPE. This Level 2 analysis would, however, use the existing BSEP PRA as the foundation for the Level 1 analysis.

The NRC's ol 'ectives for the IPE are consistent with those which had been set earlier by CP&L management, and CP&L believes that its PRA program meets all NRC requirements and goals of the Individual Plant Examination program. This report is submitted in fulfillment of the requirements of Generic Letter No. 88-20.

1.2 Plant Familiarization

The two-unit Brunswick Steam Electric Plant (BSEP) is located approximately 20 miles south of Wilmington, North Carolina, near the mouth of the Cape Fear River in Brunswick County, North Carolina (Figure 1.2-1). The two nearly identical units are General Electric BWR-4 boiling water reactors (BWRs) with steel-lined concrete Mark I containments. BSEP is owned by Carolina Power and Light Company (CP&L) and the North Carolina Eastern Municipal Power Agency. CP&L has overall responsibility to ensure that BSEP is designed, constructed, and operated without undue risk to the health and safety of the public. United Engineers and Constructors, Incorporated (UE&C) was the architect engineer, and Brown and Root, Inc. was the construction contractor. Unit 2 is rated at 790 megawatts net electrical output and began commercial operation in November 1975. Unit 1 is rated 790 megawatts and began commercial operation in March 1977. Both units are rated 2436 (2531 maximum) megawatts thermal.

The BSEP site layout is shown in Figure 1.2-2. Each unit has its own Reactor Building (RB) and Turbine Building (TB). The units share a single Control Building (CB), Diesel Generator Building (DGB), Circulating Water Intake Structure (CWIS), Service Water Intake Structure (SWIS), and certain other structures. A side view of the RB showing the drywell, wetwell, and reactor vessel is shown in Figure 1.2-3.

A reactor system heat balance is shown in Figure 1.2-4. The nuclear core converts water to steam which is passed to the turbine-generator to produce electricity. Circulation of water through the core is provided by two recirculation pumps and associated loops. Water makeup is supplied by the main feedwater system, which draws water from the condenser through the condensate system.

1.2.1 Plant Systems

1.2.1.1 Reactivity Control

The Reactor Protection System (RPS) is the primary success path for reactor SCRAM and is backed up by the Alternate Rod Insertion (ARI) system, the Recirculation Pump Trip (RPT) which reduces core power, and the two-pump Standby Liquid Control (SLC) system which injects borated water into the core in the unlikely event that the primary reactivity control systems fail.

1.2.1.2 Reactor Pressure Vessel Overpressure Protection

Seven independent safety-relief valves limit system steam pressure and discharge to the suppression pool to protect the primary system from damaging overpressure. These valves not only are opened automatically by high steam pressure, but, can also be automatically or manually opened to depressurize the primary system. In addition, four relief valves open automatically on high steam pressure without the remote safety actuation capability.

1.2.1.3 Primary System Inventory Control

Primary system coolant makeup can be provided by both high and low pressure injection systems. The high pressure systems can inject water into the primary system when the pressure is at, or near, normal operating levels, These systems are:

- A steam-driven Reactor Core Isolation Cooling (RCIC) system
- A steam-driven High Pressure Coolant Injection (HPCI) system
- The Main Feedwater System (FWS), which requires the availability of off-site power to function
- The Control Rod Drive (CRD) system which also requires off-site power to function

The low pressure systems are used to inject cooling water into the core when the primary system has been depressurized, either following a LOCA, or following a condition in which the high pressure injection pumps were unable to maintain reactor vessel level and successful manual depressurization or actuation of the Automatic Depressurization System (ADS) have been initiated. The important low pressure injection systems are:

- The Low Pressure Coolant Injection (LPCI) mode of the Residual Heat Removal (RHR) system.
- The Core Spray System (CS).
- The Condensate System (CDS), which also requires the availability of off-site AC power.
- The Service Water System (SWS) and the Fire Water System (FPS) (as specified in the Emergency Operating Procedures).

1.2.1.4 Decay Heat Removal

Core and Containment decay heat removal is provided by either:

- o The RHR system in its suppression pool cooling mode
- o Re-establishing the condenser and the condensate system
- o Containment venting (under emergency conditions only).

1.2.2 Plant Support System Structure

1.2.2.1 Power Supply and Distribution

Four off-site AC electrical power lines, aligned to two different off-site substations, connect to each BSEP unit. Standby on-site AC power is provided by four emergency diesel generators (DGs). Two DGs and their corresponding Emergency Buses are "dedicated" to each unit, but, some components are normally powered by the two Emergency Buses from the other unit. It is possible to cross-connect the 4160 V and 480 V AC emergency electrical buses between units.

Two independent sets of 125/250 Vdc batteries provide emergency DC power for each unit. These are not interconnected between the units.

1.2.2.2 Plant Cooling

A five pump, once-through, Service Water System (SWS) uses chlorinated salt water from the Cape Fear Estuary to cool important plant equipment and provide a heat sink for the Reactor Building (RBCCW) and Turbine Building Closed Cooling Water (TBCCW) systems and the RHR heat exchangers. Two Nuclear Service Water pumps provide cooling to safety-related loads, and three Conventional Service Water pumps provide cooling to balance-of-plant loads.

1.2.2.3 Interfaces between Units 1 and 2

To enhance reliability, the integrated plant design has maximized the redundancy and diversity of power and cooling systems for important plant components by providing cross-powered equipment and several inter-unit cross-ties. These are:

1. Two of the four residual heat removal pumps for each unit are powered from the other unit.

2. One of the three conventional service water pumps on each unit is powered from the opposite unit (this maintains its availability in a single unit loss of power).

3. The service water intake structure, Battery Room HVAC systems, and fire protection systems are common to both units.

4. Each diesel generator can be cooled with service water from either unit.

5. Condensate can be transferred between individual Condensate Storage tanks to maximize the available inventory for the unit which needs it.

1.2.3 Operator-Plant Interfaces

The two-unit plant has a single control room in which there are two nearly identical, but separate, control panels. Each member of the plant operating staff is assigned to a single unit and works within a single operating crew. Each operating crew stays together during both operating and training activities. Since the control room panels are nearly identical, a single simulator is able to provide equal plant fidelity for both Unit 1 and Unit 2 operating staffs.

1.2.4 Inter-Unit Comparisons for BSEP Unit 1 and BSEP Unit 2

The BSEP units are essentially identical in design, with the few exceptions briefly described below. The similarity between the designs means that modifications or operational changes which have been developed to enhance plant safety and reliability are implemented on both units, generally on a staggered schedule based on the refueling outage schedules. The design comparison between Units 1 and 2 identified two potentially significant differences:

- The turbine bypass capacity for Unit 1 is 22 percent of rated power, whereas on Unit 2 it is designed to handle 88 percent.

- A supplemental drywell cooler can supply either unit's drywell with additional cooling.

For the purposes of the PRA, these design differences were equated to the most limiting condition and a single, conservative, model was developed to represent either BSEP unit. The Drywell Coolers are not credited for lowering temperature after a transient on either unit. Review of the data developed from plant history indicated that a difference in turbine bypass capacity had an insignificant effect on the related initiating event frequencies. The ability to bypass steam to the condenser is important in events involving an Anticipated Transient Without Scram (ATWS) with a concurrent failure of the Recirculation Pump Trip function. However, the frequency for this accident sequence is very low and it is therefore not necessary to differentiate bypass capabilities between the two units.

1.3 Overall Methodology

The IPE submittal for BSEP consists of a Level 1 PRA which was analyzed with the small event tree/large fault tree methodology, and a Level 2 PRA which was analyzed with a BSEP-specific Containment Event Tree and a proprietary micro-computer version of the Source Term Code Package (STCP).

Since the existing BSEP PRA was to become the foundation for the IPE, the first task for the IPE project was one of reverification. This was to ensure both that the model reflects up-to-date plant operations and the latest advances in PRA technology, and that the assumptions made during its original development continued to remain valid.

The Level 1 event trees detail the post-initiator plant behavior for each identified initiator category and were used to investigate and predict the frequencies for possible accident sequences. The fault trees provided a logical representation of hardware and human failure characteristics which were relevant to the success or failure probabilities for each functional element in the event trees. Plant-specific databases were developed for human errors, common cause failures, component failures, and initiating events so that realistic quantification of the BSEP accident sequence frequencies could be performed.

The event trees, results and existing Level 1 information were used as much as possible during the development of the internal flooding analysis. Plant walkdowns were performed to identify potential flooding sources and spatial dependencies which affect the ways in which sprays could damage nearby equipment and water could accumulate and submerge important equipment in specific areas. The Level 1 event trees were modified to delineate the flooding core damage sequences which result from flood induced dependent failures, and to provide the basis for the quantification of their associated frequencies.

The Level 1 accident sequences with similar characteristics were grouped into key plant damage states which serve as the entry points for the Level 2 containment event trees (CET). The CETs model accident progression from the onset of core damage to the point of containment failure and release of radionuclides to the environment.

The development and quantification of these trees included the depth and detail needed to reflect current state-of-the-art knowledge of severe accident phenomena. The similarity in release characteristics (timing, energy content, magnitude, etc.) was used to group the many individual containment event tree end states, so that a single sequence could be used to represent and characterize the source term for the each release category. The release category source terms were calculated from detailed fission product transport calculations performed with the BSEP models and the proprietary version of the Source Term Code Package.

1.4 Summary of Major Findings and Insights

1.4.1 Major Findings of the Level 1 IPE

The estimated me. :ore damage frequency (CDF) for the Level 1 IPE is 2.7E-5 per year. The important accident sequences and their individual contributions to the overall estimate of CDF are briefly described below.

1.4.1.1 Station blackout accident sequences contribute 66% to the overall CDF. These sequences involve:

- successful scram following a loss of offsite power
- failure of the emergency diesel generators to start and run
- failure to recover offsite power to Unit 2 or use the Unit 1 cross-tie to restore power to the Unit 2 emergency buses

  To prevent battery depletion and consequential core damage, at least one emergency bus must be recovered within 2 hours if initial injection with HPCI or RCIC is successful, and within 30 minutes if initial injection is unsuccessful. Because control power for the switchyard breakers comes from these same DC batteries, recovery of off-site power is difficult even if it is available to be restored after battery depletion.

1.4.1.2 Transient initiated sequences which involve Loss of Decay Heat Removal contribute 30% to the overall CDF. The majority of these sequences involve:

- either loss of offsite power or closure of the MSIVs
- successful scram and injection of cooling water to the core
- loss of all three long term decay heat removal options:

1) Failure of the Residual Heat Removal in its suppression pool cooling mode.

2) Inability to re-establish the condenser as a heat sink while using the Condensate pumps to supply core cooling water.

3) Venting the containment to remove decay heat.

1.4.1.3 Accident sequences involving ATWS (3%), Transients with Loss of High Pressure Injection, (1%), LOCAs (<1%), and Interfacing LOCA (<1%) constituted the remainder of the accident sequences which were above the analytical truncation level of 1E-8 per year.

Table 1.4.1-1

Accident Types and Their Contribution to Core Damage Frequency

Accident Type	Core Damage Frequency (/yr)	Percent Contribution to total CDF
Station Blackout	1.8 E-5	66
Transient With Loss of Decay Heat Removal	8.3 E-6	30
Anticipated Transient Without Scram	7.0 E-7	3
Transient With Loss of High Pressure Injection	3.1 E-7	1
LOCA	1.6 E-7	<1
Interfacing System LOCA	3.8 E-8	<<1
TOTAL	2.7 E-5	

It is important to note that, contrary to the assumptions initially made during the Level 1 analysis, CRD injection could maintain adequate vessel level and that some of the core damage sequences involving loss of decay heat removal which are included in the results, may not resu!' in core damage. Only if the sequence also involves CRD failure, does core damage ensue. This valuable insight came from the analysis performed to determine the BSEP containment response characteristics, needed for the Level 2 analysis.

For the purposes of this IPE submittal, it was decided not to change these Level 1 sequences. The effects of changes in the success criteria for injection with the CRD pumps will be analyzed as part of the ongoing maintenance and application program planned for the "living" Brunswick PRA. This decision results in a conservative estimate of CDF.

1.4.1.4 Component Importance

The results of the Level 1 analysis included assessments of individual component importance m-.sures which can be used to provide an overall ranking of individual components. The importance measures u:-! were:

- Risk Achievement, in which individual component failure probabilities are decreased to determine the effects on the calculated results. This measure provides an indication of the maximum possible gain which can be achieved from a decrease in the unavailability of a single component.
- Risk Reduction, in which in which individual component failure probabilities are increased to determine the effects on the calculated results.
  This measure provides an indication of the maximum possible effect which will result from an increase in the unavailability of a single component.
- Fussell-Vesely importance.
  This measure provides an indication of the sensitivity of the final calculated result to the unavailability of a single component.

The results of this assessment were consistent for each importance measure and showed that the most important plant characteristics related to the unavailability and unreliability of the diesel generators following a loss of offsite power event. This was expected, because the diesel generators play a major role in each station blackout (SBO) sequence. The SBO sequences represented the single largest functional contributor to overall core damage frequency.

Components in the Instrument Air System (IAS), the service water system (SWS) and the RHR systems were also identified to be important.

1.4.2 Major Findings from the Level 2 IPE

A complete analysis of the progression of severe accidents in the Brunswick plant was performed in the IPE Level 2 analysis. Each core damage sequence identified in the Level 1 analysis was considered during performance of the Level 2 analysis. Each accident sequence was carefully examined, its functional characteristics identified and a set of key accident sequences was defined. These key accident sequences each had unique characteristics which, in total, were shown to be representative of the Level 1 sequences. Each individual key sequence could then be used to represent groups of Level 1 sequences during the performance of the accident progression, containment response and source term assessments for BSEP. A plant-specific containment event tree was developed and later quantified to provide frequency estimates for each source term.

A Brunswick-specific Containment Systems Event Tree was used to extend the Level 1 accident sequences to include the status and occurrence probabilities for events and system failures which play no role in core damage accidents, but, which are important to containment performance. Sequences generated by the combination of the Level 1 cutsets with their containment systems event tree extensions, produced the necessary linkage between the Level 1 and Level 2 analyses. This intermediate state definition was referred to as Level 1.5. Each Level 1.5 sequence with a frequency greater than 10 percent of the cutoff frequency for IPE reporting was grouped into a unique set of Brunswick plant damage states, each of which contained a group of cutsets with the same functional accident characteristics. These damage states and their associated frequencies became the starting point for the Level 2 accident progression and containment response analysis.

The plant damage states were further condensed into representative sequences which could be used to represent each plant damage state. This was done to limit the complexity of the analysis without losing important information, and was achiev. in the following way. The sequence cutsets were ordered by frequency within ach plant damage state and examined to find the single sequence which best represented the group. By focussing on the most likely sequences, the impacts from minor deviations between individual quences and the selected representative sequence were minimized. The product from this task was a limited set of Ke, Plant Damage States (KPDS) which could be used to represent the much larger number of individual accident sequence cutsets.

The frequency associated with each KPDS was calculated by summation of the frequencies for all individual sequences included in the group. The results of this condensation process are shown below

KPDS	Frequency (/yr)	Accident Sequence
IAe1	1.82E-5	Containment initially intact. High vessel pressure at the time of core melt and no water available to the core debris. rywell spray, and no venting. This state is initiated by either a sta. blackout or a transient with failure of high pressure injection.
IAd3	2.17E-7	Containment initially intact. Low vessel pressure at core melt and no water to the core debris. Venting is performed after vessel breach. This state is initiated by a transient with a loss of service water and loss of high pressu injection.
IAe3	4.65E-7	Containment initially intact. Low vessel pressure at the time of core melt and no water to the core debris. No drywell spray, and no venting. This state is similar to IAd3 except there is no venting after core melt.

EAe1	7.73E-8	Early containment breach due to failure to isolate. High vessel pressure at the time of core melt and no wa· · available to the core debris. No drywell spray, and no venting. This state is initiated by a station blackout sequence.
YBe3	3.80E-8	Interfacing ‿ystem LOCA. All equipment in RB fails. CST injected. No water to core/debris.

Accident progression was analyzed with a Brunswick specific model of the MARCH-RMA and CONTAIN codes for each of these five key p'·nt damage states. This analysis was structured to determine:

(1) The timing and magnitude of key events,
(2) The ability of the containme.. to mitigate the accident consequences
(3) The source term released to the environment.

A set of release categories was defined to characterize the range of source terms from severe accidents at Brunswick. A containment phenomenological event tree (CPET) was developed specifically for BSEP and the outcome from each path in the CPET was assigned to one of the defined release categories. The CPET provided the structure through wh ch the uncertainty associated with key accident processes was examined and the means for quantifying the frequency estimate for each release category. The CPET was quantified separately for each key accident sequence. Branching probabilities or split fractions for the CPET were derived from a best estimate, accident progression analysis and from an assessment of the uncertainties and sensitivities in the models.

The results of the Level 2 analysis provided the following frequency estimates for the five containment failure categories which encompass the accident sequence groups identified in the IPE generic letter: (Note that the total CDF value of 1.9E-5 per year reflects the credit taken in the Level 2 thermal hydraulic analyses for several Level 1 sequences that were found, as discussed in this submittal, to result in a non core damage end state.)

	Containment failure category	Frequency
I.	Intact and isolated containment	2.3E-7/yr
II.	Venting after core melt	2.1E-7/yr
III.	Containment failed late	1.6E-5/yr
IV.	Containment failed early	2.4E-6/yr
V.	Containment bypassed	3.8E-8/yr
	Total	1.9E-5/yr

The results show that about 1% of CDF is associated with an intact containment and no release. Another 1% of CDF accounts for sequences where the containment is vented after core damage. A further 85% of CDF is associated with sequences where the containment fails several hours or even days after vessel melt-through. The majority of this frequency is from KPDS IAe1 with a containment failure time of 22 hours. The time between vessel breach and containment failure is about 10 hours, which is sufficiently long to achieve a significant source term mitigation by fission product deposition inside the containment. Early containment failure occurs in accident sequences which represent about 12% of CDF. The time of early containment failure is either before or shortly after vessel melt-through, and the source terms for these sequences could be more significant. The remaining 0.2% of CDF is associated with containment bypass sequences. These sequences could also have a more significant source term.

The fraction of CDF from sequences leading to early containment failure is small and probably smaller than most BWRs, because the [illegible] melt-through failure mode at BSEP does not lead to a large source term due to the reinforced containment design. The more important question raised by these results is the basis for the relatively high fraction leading to late containment failure instead of an intact containment condition. The reason is that all sequences in KPDS IAe1 eventually lead to containment failure since there is no containment heat removal and the containment is dry, i.e. the debris is not quenched and cooled, and therefore it penetrates the drywell basemat. This results in a continuous gas and energy input into the drywell that can not be removed, and which eventually leads to containment failure.

Source term release categories and their frequencies are summarized below:

Key Release Category	Frequency (/yr)
INTACT CONTAINMENT	2.3 E-7
VENTING AFTER CORE DAMAGE	2.2 E-7
LATE CONTAINMENT FAILURE	1.6 E-5
EARLY CONTAINMENT FAILURE - LOW PRESSURE AT VESSEL BREACH	1.5 E-6
EARLY CONTAINMENT FAILURE - HIGH PRESSURE AT VESSEL BREACH	8.8 E-7
CONTAINMENT BYPASS	3.8 E-8

Each key release category with a frequency greater than 1E-6 per year is considered a potentially significant source term. For the containment bypass source term, the cutoff frequency is 1E-7 per year. Of the six key release categories, late containment failure and early containment failure with low pressure at vessel breach are considered significant.

[illegible] Category	Frequency (/yr)	Accident Sequence
Late Containment Failure	1.6 E-5	Containment initially intact, but early failure occurs at the time of vessel melt-through. High vessel pressure at the time of core melt and no water available to the core debris. No drywell spray, and no venting. This state is initiated by either a station blackout or a transient with failure of high pressure injection.
Early Containment Failure - Low Pressure at Vessel Breach	1.5 E-6	Containment initially intact. High vessel pressure at the time of core melt and no water available to the core debris. No drywell spray, and no venting. This state is initiated by either a station blackout or a transient with failure of high pressure injection.

1.4.3 Comparison with Other PRAs

1.4.3.1 Comparison of the BSEP IPE to the Original BSEP PRA

The original BSEP PRA was submitted on the plant docket in the spring of 1988. The NRC contracted INEL to review the submittal, which included a Level 1 PRA and an assessment of the effects from external events. The results from this review were published in November, 1989 in NUREG/CR-5465 [1-2]. A response to each comment and finding was prepared by the CP&L staff during the performance of the IPE and was retained in the project files. The relevant INEL and NRC comments were factored into the current PRA during its development.

The original BSEP PRA served as the foundation for the current PRA and the BSEP IPE, and has been converted to a form which can be analyzed with the EPRI/SAIC CAFTA software. This conversion greatly facilitated solving the model on a microcomputer and provided consistency with the H. B. Robinson and Shearon Harris plant PRA models. The software conversion was completed by the corporate PRA group in 1989.

In 1989, it was recognized that the original BSEP PRA would require enhancement and updating to meet both the in-house objectives and the requirements of Generic Letter 88-20 as defined in NUREG 1335. The IPE evolved into a complete rework of the original PRA. Specific enhancements are discussed in the following section on internal reviews. The work was performed primarily by the corporate PRA group during the period from late 1989 through mid-1992. Assistance was provided by Halliburton NUS (HNUS) and Risk Management Associates (RMA).

The enhanced and updated IPE model results were compared to the original PRA model results to see if any major differences existed, and if so, whether they could be attributed to specific model changes. The following table summarizes the results of this comparison:

Table 1.4.3-1

Comparison of BSEP IPE Results to Original BSEP PRA Results

Accident Type	Original PRA		IPE	
	CDF(/yr)	% Contr.	CDF(/yr)	% Contr.
Station Blackout	7.8E-6	38%	1.8E-5	66%
ATWS	9.2E-6	44%	7.0E-7	3%
Transient - Loss of Injection	2.7E-6	13%	3.1E-7	1%
Transient - Loss of DHR	8.4E-7	4%	8.3E-6	30%
Other	2E-7	1%	2E-7	<1%
TOTAL	2.1E-5		2.7E-5	

The loss of offsite power event tree was modified to reflect the most recent information on event timing and a more realistic estimate of battery depletion times. The original PRA had assumed that 5 hours were available to recover offsite power and re-establish core injection and cooling in time to prevent core damage. The IPE assumes that battery depletion will occur in two hours. After battery depletion, off-site power cannot easily be restored because the batteries provide control power for the switchyard breakers.

ATWS event frequencies have decreased significantly. The original PRA did not credit the Alternate Rod Insertion system as a backup means to scram the reactor. The probability of failure to scram decreased from 3.0E-5/yr to 4.3E-6/yr. This refinement was made based on a peer review comment from an outside consultant. In addition, operator training has emphasized response to ATWS events, and to take credit for this situation, simulator observations were used to quantify the human error probability for "failure to initiate SLC injection". This probability was reduced from 3.0E-2 to 2.7E-3. The result was a significant decrease in ATWS frequency.

The DHR model developed for the original PRA model was enhanced during the IPE process to make it more realistic. Additional support system and dependent failure modes were added and the structure of the DHR fault tree was changed to include venting and re-establishing the condenser. The changes had both positive and negative effects on the sequence frequencies, but resulted in a net increase in overall frequency. In addition, the IPE model contains new accident sequences involving internal floods that culminate in failure of DHR. The overall result was a significant increase in the contribution of loss of DHR to the overall CDF.

1.4.3.2 Comparison of the BSEP IPE to NUREG/CR-4550

A review of NUREG/CR-4550, Rev. 1, Vol. 4 was performed to identify any significant differences in design, methodologies, and results between the Peach Bottom (PB) analysis and the BSEP IPE.

The front-line systems are similar for each plant, with the following exceptions:

o ADS success criteria for Peach Bottom is 3/5 valves, whereas for BSEP it is 2/7 valves. BSEP criteria were based on thermal hydraulic studies using the MARCH-RMA code.

o Peach Bottom assumed 15 hours for battery depletion time, whereas the BSEP analysis assumed battery depletion within 2 hours if offsite power is not restored. The BSEP assumption is based on a plant-specific Station Blackout coping analysis.

o The Peach Bottom analysis included recovery events for restoration of the batteries and diesel generators which BSEP did not credit because of the low success likelihood in the short period of time available before battery depletion.

These plant differences have a great effect on station blackout sequences and their associated frequencies.

The Peach Bottom analysis (NUREG/CR-4550 Rev. 0) indicates that water level control is not needed during an ATWS to prevent suppression pool overheating. This is at variance with the BSEP model which had been originally developed for the BSEP PRA and had not been changed to reflect the changes to SLC which had been made to meet the ATWS rule, (10CFR 50.62). Though the SLCS design at BSEP has since changed to 2 pump operation, the PRA model was not changed because it was felt to fairly represent the actual expected conditions if one SLC pump were to fail. The difference in success criteria means that there is less time available for the operator to initiate SLC in the single pump model than if two pumps were credited, and simulator observations were used to verify the operators' timely response during an ATWS event which requires SLC injection.

The "Failure to Scram" model for BSEP was updated based on an analysis which included Alternate Rod Insertion and which separated Reactor Protection System (RPS) failures into mechanical and electrical failures.

The Peach Bottom containment is a typical Mark I steel shell design whereas the BSEP containment has a steel-lined concrete containment. This difference has a large impact on containment failure characteristics. The typical failure mode for a steel-shell Mark I containment is a liner rupture at the wetwell airspace or the knuckle area in the drywell. The predicted failure mode at BSEP is flange separation at the drywell head.

A summary table of these differences is provided in Table 1.4.3-2.

Table 1.4.3-2

Comparison of Functional Features of BSEP and PB (NUREG 4550)

System/Function	BSEP	PB	Effect
DC Power	2 hour depletion	15 hour depletion	Significant. Insufficient time to recover DGs, or Batteries.
AC Power	no credit for recovering DGs	recovery of DGs included	Significant
Automatic Depressurization	2 of 7 SRVs for success	3 of 5 SRVs for success	Not a significant effect on timing or failure to depressurize.
ATWS	ARI included in Scram failure probability Plant-specific data on failure to initiate SLC		Significant credit is taken for plant specific response to ATWS events.
Primary Containment	concrete with steel liner	steel shell	Higher pressure to failure and different release location.

1.4.4 Insights

The results of the IPE indicate that the most important contributors to core damage frequency and the frequency of a release from containment originate with a station blackout. The factors which influence the frequency for this important functional sequence have been discussed earlier in this section, but, in summary are:

- o The frequency of occurrence for loss of offsite power events
- o Diesel generator unavailability and reliability
- o Short depletion times (2 hours) for the station batteries
- o The difficulty in restoring offsite power after the batteries are depleted, because the switchyard breakers depend upon the same batteries for breaker control power.

The frequency of loss of offsite power for the site and the unavailability of the diesel generators are within normal industry standards. This means that the available options which could reduce the frequency for station blackout sequences appear to require a change in the plant reliability structure. Viable options appear to be:

- o Increasing the overall reliability of the on-site power generating system, i.e., increasing the number of diesel generators or providing an auxiliary power source which is capable of recharging the batteries so that off-site power can be re-established when the electric grid is restored.
- o Changing the power source for the switchyard breaker controls so that recovery of off-site power can be accomplished even if the emergency source of DC power from the station batteries is unavailable.
- o Increasing the battery depletion times by providing enhanced battery load shedding capability.

Though not as important as station blackout, loss of DHR is also important to risk at BSEP. To reduce the frequency of these sequences:

- o Increase the reliability of containment venting
- o Ensure long term CST refill capability so that injection can operate for a long time and delay core damage until there is a high likelihood that the DHR systems can be restored.

Several of the options detailed above were identified independent of the IPE analysis and are in various stages of design and implementation. These specific improvements include:

- o Installation of a fifth diesel generator
- o Remote capability to cross-tie emergency buses to increase plant power system flexibility and reliability.
- o Establish a separate source of power for the switchyard breaker control system so that the breakers can be remotely operated following depletion of the station batteries, and allow restoration of off-site power to the unit as soon as the grid and the transmission lines have been re-energized.
- o A hardened wetwell vent which will reduce the actions necessary to establish a vent path from the containment.

Details on these modifications can be found in Section 6. While it is not possible to model the planned configuration until design details are complete, it is expected that the overall effect of these modifications will be to reduce CDF due to station blackout to less than 1E-5 per year. This is based on sensitivity studies performed to evaluate the benefits of the fifth diesel and other modifications. In addition, the ability to mitigate a long term station blackout event will significantly reduce the overall core damage frequency and the probability of containment failure.

The BSEP response to the generic Containment Performance Issues and Decay Heat Removal issues are addressed in Section 3.4. The conclusions of both assessments are that the present plant configuration and already low core damage frequencies, along with modifications already underway will adequately address these issues.

1.4.5 Regulatory Concerns and Their Effects on the IPE

On July 2, 1992, the Brunswick plant was placed on the NRC "Watch List" with a Category 2 rating. The areas of NRC concern were primarily related to lack of administrative control of plant configuration and plant material condition. Studies were performed to determine how these regulatory concerns might affect the PRA model to ensure that the model reflects the as-built, as-operated condition of the plant.

The PRA model includes best estimates of plant component unavailabilities, test and maintenance unavailabilities, probabilities of failure to restore equipment to operable status following test and maintenance, and human errors. Though the concerns about plant work control do not directly affect the use of, or training in, plant operating procedures, a sensitivity study was performed on the human error probabilities which have been used in the PRA. The results showed that even if each human error probability (HEP) which was less than 1E-2 was increased by an order of magnitude, and each HEP at 1E-2 and above was increased by a factor of 3, the overall core damage frequency would increase to 1.5 E-4 per year. This would represent the upper bound of any uncertainty in the characterization of human errors at BSEP.

As another means to assess the effects of current regulatory concerns, two studies were performed to determine the sensitivity of the component and maintenance unavailabilities on the CDF. In the first study, generic data for major BSEP components was replaced with corresponding data from Peach Bottom (NUREG/CR 4550). The resulting CDF was less than 3% higher, which leads to the conclusion that uncertainties in the failure behavior of plant components is relatively insensitive to failure rate estimates. The second study replaced BSEP maintenance unavailability data with Peach Bottom data again. The results of this study were a 35% decrease in CDF using the generic maintenance data from the Peach Bottom study. Therefore, the maintenance unavailability data currently used in the BSEP IPE is conservative and any uncertainties would be expected to be well within the range of the current results.

Another class of failures that could be affected, post maintenance failure to restore errors, are typically one or two orders of magnitude lower than normal test and maintenance unavailabilities and active component failure modes and thus are very insensitive to uncertainties in data.

In May of this year, BSEP plant staff requested the PRA group to review a listing of "temporary conditions" to assess their cumulative impact on safety. The conditions consisted of clearances greater than 3 months old, temporary changes to Engineering Evaluation Reports, Equipment Out of Service logs, and Jumpers and Lifted Leads logs. The PRA group's analysis identified three temporary conditions that could potentially impact plant safety. The conditions involved the Instrument Air compressors and two cases of RHR valve discrepancies. After discussions with the system engineers on the details of each condition, the conclusion was that none of the three posed a significant increase in risk. Most of the temporary conditions could be characterized as either not affecting the PRA model, or as minor problems that did not affect the operability of systems. There was no indication of common cause or synergistic effects based on the cumulative impact of temporary conditions.

Another request was made by the BSEP plant to evaluate the effect of many structural deficiencies referred to as Short Term Structural Integrity Items (STSI). The STSI list consisted of many conditions in the plant where structural items did not meet the design criteria and had been "short term qualified" based on analysis of their as-found condition. This list included the condition of the Diesel Generator Building walls which were missing several bolts and the Service Water Pump shafts. These conditions did not affect the internal events PRA, but they would affect a seismic PRA. The effect of several piping supports failing due to a seismic event or a water hammer event was assessed by performing plant walkdowns of the STSI supports. No cases were found where several consecutive supports would fail or "unzip" in these conditions. The assessment concluded that the overall affect was an increase in core damage risk, but relatively small and well below the target safety goal of 1E-4/yr.

Even though the recent criticisms voiced by the NRC are extremely important and are being actively addressed by CP&L, they do not have a significant effect on the PRA results or the character of the accident scenarios which have been identified as dominant. The sensitivity studies conclude that the current results of the PRA are valid and the insights drawn from these results are useful guidance for plant management.

1.5 References for Section 1

1-1 Brunswick Steam Electric Plant Probabilistic Risk Assessment. Southport, North Carolina, Carolina Power & Light Company/EI International, April, 1988.

1-2 Review of the Brunswick Steam Electric Plant Probabilistic Risk Assessment. Washington, DC: US Nuclear Regulatory Commission, NUREG/CR-5465, November, 1989.

1-3 Brunswick Decay Heat Removal Probabilistic Safety Study. Impell Corporation, NSAC-83, October, 1985

1-4 Program for Assessing Risk and Evaluating Proposed Modifications at Brunswick Steam Electric Plant. Walnut Creek, California: Impell Corporation, November, 1985. Report No. 01-1320-1421, Rev. 0.