BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2,

INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(BACK-END)

Enclosure 3

SCIE-NRC-229-94

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BRUNSWICK UNITS 1 AND 2 TECHNICAL EVALUATION REPORT ON THE INDIVIDUAL PLANT EXAMINATION BACK-END ANALYSIS

Hanry A. Wagage

Prepared for the U.S. Nuclear Regulatory Commission
Under Contract NRC-04-91-068-29
May 1995

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E. Executive Summary

SCIENTECH, Inc., performed a submittal-only review of the back-end portion of the Carolina Power and Light Company (CP&L) Individual Plant Examination (IPE) of the Brunswick Steam Electric Plant (BSEP) Units 1 and 2.

E.1 Plant Characterization

The two units of BSEP are nearly identical and are General Electric BWR-4 boiling water reactors (BWRs) with steel-lined concrete Mark I containments. United Engineers and Constructors, Incorporated, was the architect engineer, and Brown and Root, Inc., was the construction contractor. Each unit has its own Reactor Building and Turbine Building. The units share a single Control Building, Diesel Generator Building, Circulating Water Intake Structure, Service Water Intake Structure, and certain other structures.

The BSEP Units 1 and 2 began commercial operations in March 1977 and November 1975, respectively. Each unit was designed to produce a reactor thermal power output of 2,436 MW with a gross electric power output of 790 MW.

E.2 Licensee IPE Process

The CP&L corporate probabilistic risk assessment (PRA) group performed the back-end analysis with contractor assistance from a staff member from Risk Management Associates. CP&L personnel reviewed the IPE and became cognizant of its results and implications.

The few differences in design between the two BSEP units were taken into consideration in the IPE, either by modeling with the most limiting condition or by not taking credit for features that differed between the two. The BSEP IPE team made use of a model of Unit 2 as it existed as of January 1, 1992. To confirm the "as-built" plant configuration, the IPE team relied mainly on the information available in a videotape and a three-dimensional, computer-aided design model of the containment and did not perform a specific back-end walkdown. To confirm the "as-operated" plant conditions, the IPE team discussed and "talked through" important accident scenarios with the operators and the operator training staff.

E.3 Back-End Analysis

Based on the front-end analysis, the team calculated an overall core damage frequency (CDF) of 2.7E-5 per reactor year from internal events. The largest CDF initiator would be a loss of offsite power/station blackout (LOP/SBO) accident (66%), followed by a transient involving a loss of decay heat removal (30%). The remaining contributors to CDF would be an anticipated transient without scram (3%), transients with loss of high-pressure injection (1%), a loss of coolant accident (LOCA) (< 1 percent), and interfacing systems LOCA (< 1%). The back-end thermal-hydraulic analysis showed that several sequences assumed to lead to core damage during the front-end analysis would not actually do so. After taking credit for this finding, the team reduced the overall CDF to a value of 1.9E-5 per reactor year. The back-end analysis that SCIENTECH reviewed was based on this CDF. (The IPE team later calculated a lower CDF of 1.1E-5 per reactor year as a result of model changes that were made in an update process of the IPE model.)

In performing the back-end analysis of the BSEP IPE, the CP&L team used a BSEP-specific containment event tree (CET) and a proprietary microcomputer version of the Source Term Code Package. The IPE team grouped the front-end accident sequences with similar characteristics into key plant damage states (KPDS) which were used as the entry points to the CETs. The CETs modeled accident sequences from the onset of core damage to failure of containment and release of radionuclides to the environment. The team used the following computer codes for the corresponding analyses: BWRSAR for in-vessel melt phase, TRAPMELT3 for in-vessel fission product behavior, and CONTAIN for ex-vessel fission product behavior and containment response analysis. CONTAIN included CORCON to model corium-concrete interactions (CCIs) and VANESA to model aerosol and fission product generation during CCIs.

The IPE team assumed that all core damage sequences progressed by breaching the reactor pressure vessel (RPV) and did not take any credit for the in-vessel recovery. Unlike many other Mark I containments, the BSEP containment was considered unlikely to fail as a result of drywell liner melt-though, primarily because the liner was backed by and anchored into reinforced concrete. The unlikelihood of liner melt-though lowers the conditional probability of early containment failure. Because the BSEP containment was inert, hydrogen combustion would not contribute to containment failure. Early containment failure would result from containment overpressurization during vessel failure and vessel thrust forces generated during high-pressure vessel failure.

The unavailability of water on the pedestal floor to cool debris and the unavailability of containment heat removal systems would result in late containment failure as a result of containment overpressure and corium-concrete interactions. The submittal describes the phenomenological uncertainties of accident progression.

The back-end analysis showed that fission product release occurred in all of the accident sequences, except those sequences that amounted to 1.2% of the CDF, where the containment remained intact. The largest contributor to the fission product release frequency postulated was late containment failure from overpressure (85.1% of the CDF) followed by early containment failure from overpressure and vessel thrust forces (12.4% of the CDF). For every core damage event, containment venting would amount to 2.1% of the CDF and containment bypass would amount to 0.2% of the CDF.

E.4 Containment Performance Improvements (CPI)

After considering the Containment Performance Improvement (CPI) Program recommendations, the IPE team concluded that, because of the BSEP design and the modifications that BSEP had committed to make, no further actions were necessary to resolve the Mark I containment issues. These modifications included hardened wetwell vents, a fifth diesel generator with dedicated switchyard batteries, and a remotely operated cross-tie.

Between the time the CP&L IPE report was issued in August 1992 and a response was made to the NRC's RAI in February 1995, the hardened vent modifications for both units had been made. [1, 2] However, also during that same time period, the utility decided not to add a fifth diesel generator or the switchyard batteries. The reasons for this decision according to the submittal were that (1) the ability to cope with the loss of DC power had improved significantly because of added provisions for DC load shedding and (2) because in many cases DC power was available for 4 or 5 hours, significantly increasing the availability of depressurization.

E.5 Vulnerabilities and Plant Improvements

The BSEP IPE submittal notes the following with respect to severe accident vulnerability:

> The team used the NUMARC Severe Accident Closure Guidelines (NUMARC 91-04) 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. The team reviewed the cutsets from the dominant accident sequences (above 1E-7/yr) and any associated component importance measures to select candidate items for comparison with the established criteria.

The NUMARC guidance is not quantitative and therefore is not measurable. By comparing the important issues identified in the IPE analysis (which are not listed in the submittal) and the list of plant modifications committed for installation at BSEP, the team concluded that resolution of most of the issues raised in IPE was well underway.

E.6 Observations

Based on its review, SCIENTECH noted that the BSEP IPE team used state-of-the-art computer codes to calculate corium-concrete interactions (CORCON) and fission product behavior (VANESA) in the back-end analysis.

SCIENTECH also noted the following strengths in the BSEP IPE back-end analysis:

- The containment phenomenological event tree (CPET) top-events quantification is described in detail.

- The submittal provides extensive details of results predicted by BWRSAR and CONTAIN for KPDSs.

- CP&L addressed phenomenological uncertainties associated with accident progression by using containment system event tree and CPET top events.

- CP&L gained experience in understanding the plant response to severe accidents by performing and reviewing the IPE.

However, the following two issues are outstanding at the time of completing this TER. The NRC is in the process of their resolution with the utility.

- The possibility of high temperature failure of the cables to the safety/relief valve pilot valve solenoids which were the only pieces of equipment located in the containment that were important for the purposes of the Level 2 analysis.

- The reasons for the low likelihood of the drywell liner meltthrough leading to early containment failures.

1. INTRODUCTION

1.1 Review Process

This technical evaluation report (TER) documents the results of the SCIENTECH review of the back-end portion of the Brunswick Steam Electric Plant Units 1 and 2 Individual Plant Examination (IPE) submittal. [1, 2] This technical evaluation report complies with the requirements for IPE back-end reviews of the U.S. Nuclear Regulatory Commission (NRC) in its contractor task orders, and adopts the NRC review objectives, which include the following:

- To help NRC staff determine if the IPE submittal provides the level of detail requested in the "Submittal Guidance Document," NUREG-1335
- To help NRC staff assess if the IPE submittal meets the intent of Generic Letter 88-20
- To complete the IPE Evaluation Data Summary Sheet

A draft TER for the back-end portion of the Brunswick IPE submittal was submitted by SCIENTECH to NRC in July 1994. Based in part on this draft submittal, the NRC staff submitted a Request for Additional Information (RAI) to Carolina Power & Light Company (CP&L) on November 9, 1994. CP&L responded to the RAI in a document dated February 27, 1995. [2] This final TER is based on the original submittal and the response to the RAI.

Section 2 of the TER summarizes SCIENTECH's review and briefly describes the Brunswick IPE submittal as it pertains to the work requirements outlined in the contractor task order. Each portion of Section 2 corresponds to a specific work requirement. Section 2 also outlines the insights gained, plant improvements identified, and utility commitments made as a result of the IPE. Section 3 presents SCIENTECH's overall observations and conclusions. References are given in Section 4. The Appendix contains an IPE evaluation and data summary sheet.

1.2 Plant Characterization

The two-unit BSEP is located 20 miles south of Wilmington, near the mouth of the Cape Fear River in Brunswick County, North Carolina. The two units are nearly identical and are General Electric BWR-4 boiling water reactors (BWRs) with steel-lined concrete Mark I containments. United Engineers and Constructors, Incorporated, was the architect engineer, and Brown and Root, Inc., was the construction contractor. Each unit has its own Reactor Building and Turbine Building. The units share a single Control Building, Diesel Generator Building, Circulating Water Intake Structure, Service Water Intake Structure, and certain other structures. BSEP is jointly owned by CP&L and the North Carolina Eastern Municipal Power Agency. CP&L is responsible to ensure that BSEP is designed, constructed, and operated without undue risk to the health and safety of the public.

The BSEP Units 1 and 2 began commercial operations in March 1977 and November 1975, respectively. Each unit was designed to produce a reactor thermal power output of 2,436 MW with a gross electric power output of 790 MW. The BSEP IPE team made use of a model of Unit 2 as it existed as of January 1, 1992.

2. TECHNICAL REVIEW

2.1 Licensee IPE Process

This section is structured in accordance with Task Order Subtask 1.

2.1.1 Completeness and Methodology.

The submittal appears to be complete in accordance with the level of detail requested in NUREG-1335 and appears to meet the NRC sequence selection screening criteria described in Generic Letter 88-20.

The IPE methodology used is described clearly. The approach followed is consistent with the basic tenets of Generic Letter GL 88-20, Appendix 1.

In performing the back-end analysis of the BSEP IPE, the CP&L team used a BSEP-specific CET and a proprietary microcomputer version of the Source Term Code Package. The IPE team grouped the front-end accident sequences with similar characteristics into key plant damage states (KPDSs) which were used as the entry points to the CETs. The CETs modeled accident sequences from the onset of core damage to failure of containment and release of radionuclides to the environment. The team used the following computer codes to perform several analyses: BWRSAR for the in-vessel melt phase, TRAPMELT3 for in-vessel fission product behavior, and CONTAIN for ex-vessel fission product behavior and containment response. CONTAIN included CORCON to model CCIs and VANESA to model aerosol and fission product generation during CCIs.

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

Although the BSEP units are essentially of the same design, a comparison made by the IPE team did lead to the identification of two differences between them (Section 1.2.4, pages 1.2.4 and 1.2.5):

- The turbine bypass capacity for Unit 1 was 22% of rated power; for Unit 2 it was 88%.
- A supplemental drywell cooler could supply either unit's drywell with additional cooling. (In what way this presents a difference between the units was not made clear.)

The IPE team accounted for these differences by assuming the most limiting condition and developing a single, conservative model to represent each BSEP unit, based on a model of Unit 2 as it existed on January 1, 1992. Furthermore, the team did not take credit for features that differed between the two, including the unidentified ones involving the drywell coolers whose function is to lower temperature after a transient.

The bypass capacity of the turbine was considered important in events involving an anticipated transient without scram (ATWS) with concurrent failure of the recirculation pump trip function. However, the team concluded that the frequency of this accident sequence was too low to warrant a recognition of the difference between the bypass capacities between the two units.

To confirm the "as-built" plant configuration, the IPE team relied mainly on the information available in a videotape and a three-dimensional, computer-aided design model of the containment and did not perform a specific back-end walkdown. To confirm the "as operated" plant conditions, the IPE team "talked through" important accident scenarios with the operators and the operator training staff.

2.1.3 Licensee Participation and Peer Review.

The CP&L corporate PRA group performed the back-end analysis with contractor assistance from a staff member from Risk Management Associates.

A CP&L team reviewed the findings of the BSEP IPE, which included the postulated results of severe accident scenarios and recommended to the nuclear senior management appropriate actions to mitigate the consequences. The team concluded that the plant modifications, which were in progress, would be sufficient to address the principal contributors to severe accident risk that the BSEP IPE team identified. A staff member from Science Applications International Corporation reviewed the IPE to confirm that the methodology, results, and conclusions of the analysis were reasonable.

By performing and reviewing the IPE, BSEP personnel reported that they had gained insights into plant behavior under severe accident conditions.

2.2 Containment Analysis/Characterization

2.2.1 Front-end Back-end Dependencies.

Based on the front-end analysis, the IPE team calculated an overall core damage frequency of 2.7E-5 per reactor year from internal events. The largest CDF initiator would be a loss of offsite power/station blackout (LOP/SBO) accident sequences (66%) which would involve 1) successful scram following a loss of offsite power, 2) failure of the emergency diesel generators to start and run, and 3) failure to recover offsite power to Unit 2 or failure to use the Unit 1 cross-tie to restore power to the Unit 2 emergency buses.

The second largest CDF initiator would be a transient accident sequence involving a loss of decay heat removal (30%). In most cases, such a sequence would involve 1) either loss of offsite power or closure of the main steam isolation valves, 2) successful scram and injection of cooling water to the core, and 3) loss of all three long-term decay heat removal options (failure of the residual heat removal in its suppression pool cooling mode, inability to re-establish the condenser as a heat sink while using the condensate pumps to supply core cooling water, and inability to vent the containment to remove decay heat). The remaining contributors to CDF above the analytical truncation level of 1E-8 per reactor year would be the sequences involving the following: an anticipated transient without scram (3%), transients with loss of high-pressure injection (1%), a loss of coolant accident (LOCA) (< 1%), and interfacing systems LOCA (<< 1%).

The back-end thermal-hydraulic analysis showed that, contrary to the assumptions initially made during the Level 1 analysis, control rod drive (CRD) system 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, might not actually result in core damage. Only if a sequence also involved CRD system failure would core damage occur. (Section 1.4.1.3, page 1.4.3) After taking credit for this finding, the team reduced the overall CDF to a value of 1.9E-5 per reactor year.

The front-end model defined core damage bins (CDBs), which were to collect all cutsets according to the characteristics of each bin. To define plant damage states (PDSs), these CDBs had to be designated based on the state of the containment systems (e.g., drywell spray, suppression pool cooling, containment isolation, and reactor building isolation). The IPE team considered the following issues of importance to the BSEP Level 2 analysis in defining the PDSs:

- Whether an accident scenario leads to containment bypass
- The status of containment isolation
- Transient type: LOCA (large, small, or medium), transient
- Whether the event involves station blackout
- If and when power is recovered
- Whether depressurization is feasible or has occurred
- Whether containment sprays are available during the scenario for energy removal and containment atmosphere scrubbing
- Whether suppression pool cooling is available during the scenario
- The reactor coolant system (RCS) pressure at the time of vessel failure
- The status of in-vessel injection (deadheaded, failed, recovered)

The IPE team developed a containment systems event tree (CSET) with fault tree models of the containment systems. The CSET, which is shown in Figure 4.3-1, page 4.3.18 of the submittal, consists of the following top events:

- Containment isolation and system integrity status (CI)
- Availability of drywell spray (DS)
- Containment venting after core melt (CV)
- Reactor building isolation status (RI)

The PDSs were the endstates of the CSET and they were used as input to the containment phenomenological event tree (CPET) to analyze containment performance during severe accidents. This was a sound methodology to use in the analysis of front-end back-end dependencies.

2.2.2 Sequences with Significant Probabilities.

The IPE team characterized the BSEP containment performance using a plant-specific CPET to model the containment responses expected to follow the occurrence of each important PDS in the Level 1 analysis by considering phenomenological processes, containment conditions, and containment failure modes that could occur during severe accidents. The CPET was developed with the capacity of quantifying each PDS that represented a unique accident progression start-point with respect to the CPET. The IPE team considered the following important phenomenological events in developing BSEP CPETs:

- In-vessel debris coolability
- Ex-vessel debris coolability
- Dispersion of debris from the cavity area
- Early containment failure mechanisms:
  - Direct containment heating
  - Hydrogen burns
  - Steam explosions/spikes
- Late containment failure mechanisms:
  - Hydrogen burns
  - Basemat failure
  - Noncondensible gas generation
  - Steam production
- Containment failure modes/location:
  - Overpressure
  - Overtemperature
  - Missiles
  - Leakage at penetrations
- Events occurring in the reactor building
  - Hydrogen burn
  - Failure caused by containment failure.

The BSEP CPET consisted of 14 top events as follows (excluding the KPDS entry state):

Five events were postulated to address questions relevant to the time, from the onset of core damage to vessel breach, if in-vessel recovery did not occur:

- Top Event 2 - Debris quenched in vessel (DQ)
- Top Event 3 - Vessel depressurized before vessel breach in high-pressure melt scenarios (DP)
- Top Event 4 - No vessel breach (VB)
- Top Event 5 - Containment intact before vessel breach (FB)
- Top Event 6 - Suppression pool not bypassed before vessel breach (SB)

Five events were postulated to address phenomena that could occur during and shortly after vessel breach (blowdown, debris entrainment effects, or ex-vessel steam explosions):

- Top Event 7 - Containment intact after vessel breach (FA)
- Top Event 8 - Containment systems continue to operate (CS)
- Top Event 9 - No fission products released into reactor building due to drywell liner melt-through (LM)
- Top Event 10 - Debris quenched and cooled in drywell (DC)
- Top Event 11 - Suppression pool not bypassed late (SA)

Three events were postulated to address long-term containment response (including prevention of containment failure by debris cooling and containment heat removal by venting):

- Top Event 12 - Containment vented after vessel melt-through (TV)
- Top Event 13 - Containment intact late (FL)
- Top Event 14 - Leak containment failure mode (LK)

One event was postulated to address reactor building effectiveness in mitigating source term release after containment failure:

- Top Event 15 - Fission product scrubbing in reactor building is effective (BE).

In Table 3.4-1, page 3.4.1 of the submittal, four of the severe accident sequences in the BEEP IPE are shown to have met the Generic Letter 88-20 screening criteria. These sequences, which constituted a total frequency of 86% of the total CDF, are described in Section 3.4.1, pages 3.4.2 through 3.4.4, of the submittal. The BSEP IPE appears to have met the sequence selection criteria, as outlined in Appendix 2 to the Generic Letter 88-20.

2.2.3 Failure Modes and Timing.

Table 4.1-1, page 4.1.4 of the submittal, lists important containment data including the following:

Material:	Reinforced concrete with steel liner
Design pressure (psig):	62
Drywell:	
Free volume (ft^3):	164,100
Design temperature (°F):	300
Wetwell:	
Minimum free volume (ft^3):	124,000
Minimum water volume (ft^3):	87,600
Design temperature (°F):	220

A CP&L contractor, EQE Engineering Consultants, analyzed potential containment failure modes for BSEP. [3] EQE found five failure locations that would result in failure areas large enough to release the contents of the containment atmosphere into the reactor building. Two modes (failure of the torus and of the drywell shell) were structural failures with large leak areas that would release the contents of the containment atmosphere in a short time. The remaining three modes (failures of the head flange, vent bellows, and personnel airlock) were the results of containment leaks (leak before break) with areas sufficient to terminate further containment pressurization and thus prevent structural failure. For each of the five failure modes, EQE calculated the failure pressure with associated uncertainty for containment temperature to range from 300°F to 700°F. (See Table 4.4-1, page 4.4.14 of the submittal.) The IPE team used the CONTAIN computer code to calculate failure timing for accident sequences.

Section 4.4.2, page 4.4.13 of the submittal, notes that the electrical penetration seal leakage could have started about the same time as the drywell head leak started. However, the electrical penetration leak would be negligible compared with the drywell head leak and, therefore, a BSEP containment failure would not be sensitive to thermal degradation of electrical penetrations.

2.2.4 Containment Isolation Failure.

CPET Top Event 5, "Containment intact before vessel breach (FB)," addressed pre-existing leak paths due to isolation failures defined by the PDS. CSET Top Event "Containment Isolation (CI)," was studied to analyze isolation failures. After identifying 245 equipment penetrations into the BSEP primary containment, the IPE team selected six penetrations using criteria the team defined. The IPE submittal notes that, because the BSEP containment was nitrogen-inerted and was monitored, isolation failure would be unlikely.

The KPDS, "EAe1," which has a CDF of 7.7E-8 per reactor year (0.0041 per core damage), represented containment isolation failures.

2.2.5 System/Human Responses.

Apart from torus venting, the IPE team considered no other back-end operator actions because they were either considered during front-end analysis or were not proceduralized. The following are the CPET top events that were used to analyze operator actions:

- Top Event VB - No vessel breach: No credit was taken for recovering makeup water because such procedures or guidelines did not exist.

- Top Event CS - Containment systems continued to operate or were recovered: No credit was taken for recovering failed containment systems after core uncovery because such procedures or guidelines did not exist.

- Top Event TV - Torus vented after vessel melt-through: in the front-end analysis, it was assumed that torus venting would prevent core damage in scenarios where vessel makeup was available but torus heat removal was not. Front-end analysis showed that successful venting did prevent core damage and therefore in the back-end analysis torus venting was not considered in the sequences where vessel makeup was available. For those sequences not considered in the front-end analysis because no vessel makeup was available, the back-end analysis assumed venting would occur after core damage. Success or failure of this top event was actually defined by PDS in the CSET Top Event "Containment venting post core melt (CV)." KPDS, "IAd3," which had a CDF frequency of 2.2E-7 (0.012 per core damage) had 100% success in venting. No other KPDS involved back-end venting.

2.2.6 Radionuclide Release Characterization.

Section 4.5.3, pages 4.5-13 through 45.17 of the submittal, describes the BSEP source term event tree (STET) which defined the release categories. Figure 4.5.2, page 4.5-19, illustrates the STET, which consists of the following seven top events:

- Top Event 1: In-vessel recovery

- Top Event 2: Drywell spray available

- Top Event 3: RCS pressure at vessel breach

- Top Event 4: Containment failure

- Top Event 5: Containment failure time
- Top Event 6: Containment failure mode
- Top Event 7: Suppression pool scrubbing
- Top Event 8: Reactor building mitigation.

The success or failure of the above top events was defined either by a PDS group or the status of certain CPET top events. Although the STET defined 66 release categories, the significant releases were in 15 categories that were condensed into the following five risk-dominant key release categories (KRCs): intact containment, venting after core damage, late containment failure, early containment failure with low pressure at vessel breach, early containment failure with high pressure at vessel breach, and containment bypass. KRC definitions and frequency of KRCs appear in Table 4.7-9, page 4.7.32 of the submittal. The release magnitudes of these KRCs are given in Tables 4.6.2-1a through 4.6.2-3, pages 4.6.200 through 4.6.205 of the submittal.

Generic Letter 88-20 states that the following should be reported:

> Any functional sequence that has a core damage frequency greater than $1x10^{-6}$ per reactor year and that leads to containment failure which can result in a radioactive release magnitude greater than or equal to BWR-3 or PWR-4 release categories of WASH-1400.

The IPE notes that none of the accident sequences met the above reporting requirement. (See Section 3.4-1, page 3.4.1.) The BSEP treatment of radionuclide release characterization appears to be complete.

2.3 Accident Progression and Containment Performance Analysis

2.3.1 Severe Accident Progression.

The BSEP IPE team addressed the sensitivities in the accident progression and CPET results due to phenomenological uncertainties by using the results of previous studies, or by analyzing phenomena using CSET or CPET top events (Section 4.4.2, pages 4.4.10 through 4.4.13).

The IPE team noted the conclusion of the NRC's steam explosion review group that the probability of in-vessel steam explosions occurring was very low, i.e., on the order of 1E-4. If this event were postulated to occur at this probability, the frequency of the resulting early containment failure would be a factor of 1,000 lower than the frequency of the dominant early containment failure sequence. Therefore, the team concluded that the BSEP Level 2 results were not sensitive to assumptions about the probability of alpha-mode containment failures. Because none of the BSEP key PDSs had a wet drywell at the time of vessel breach, the team concluded that ex-vessel steam explosions were not a concern at BSEP.

The team analyzed the following phenomena using CSET or CPET top events:

- Direct containment bypass
- Failure to isolate containment
- Containment overpressurization
- Combustion processes (The team considered negligible the contribution to containment failure from combustion processes because the BSEP containment was nitrogen-inerted. Hydrogen burns in the reactor building were analyzed to find the effect on the source term. Hydrogen burns would structurally damage the reactor building, thus diminishing its effectiveness for fission product scrubbing.)
- Core concrete interaction
- Blowdown forces
- Liner melt-through
- Thermal attack on containment penetrations

Unlike many other Mark I containments, the BSEP containment was considered unlikely to fail as a result of drywell liner melt-though, primarily because the liner was backed by and anchored into reinforced concrete. This unlikelihood of liner melt-though lowers the conditional probability of early containment failure.

2.3.2 Dominant Contributors: Consistency with IPE Insights.

Table 1 of this report shows the results of SCIENTECH's comparison of the dominant contributors to the BSEP conditional containment failure probability with the NUREG-1150 PRA study results for Peach Bottom and other IPE results.

Compared with other plants, early containment failure would be less likely mainly because liner failure would not lead to containment failure: the liner is backed by a reinforced concrete wall. However, a relatively high fraction of accident sequences would lead to late containment failure. The reason is that all sequences in the KPDS "IAe1," which contributed to 96% of the total CDF (1.80E-5 per reactor year), eventually led to containment failure because no containment heat removal was available and the containment remained dry, (i.e., the debris was not quenched and cooled, thus core-concrete interactions continued generating gas and energy that led to containment failure by overpressurization).

Table 1. Containment Failure as a Percentage of Total CDF: Brunswick IPE Results Compared with the Peach Bottom NUREG-1150 PRA and Other IPE Results

Study	CDF (per rx yr)	Early Failure	Late Failure	Bypass	Intact w/o Vessel Breach	Intact w/ Vessel Breach
Peach Bottom/ NUREG-1150	4.5E-6	56	16	na	10	18
Fitzpatrick IPE	1.9E-6	60	26	na	11	3
Oyster Creek IPE	3.2E-6	16	26	7	51	0
Browns Ferry IPE	4.8E-5	46	26	na	25	3
Duane Arnold IPE	7.8E-6	47	32	0	na	21
Brunswick IPE	1.9E-5	13.5*	85	0.2	0	1.2

*Includes 1.1% (of total CDF) of containment vent after core damage

2.3.3 Characterization of Containment Performance.

In order to quantify CPET top events, the IPE team used information from the following:

- Past PRAs and published research reports
- Parametric information gained from the performance of BSEP plant-specific, deterministic analyses using the Source Term Code Package
- A containment strength assessment that provided the fragility of the containment as a function of internal temperatures and pressures

Section 4.7.3, pages 4.7.4 through 4.7.19 of the submittal, describes the quantification of top events. Table 4.7-3, page 4.7.27, lists split fractions used for KPDSs. It appears that CP&L characterization of the BSEP containment performance is complete.

2.3.4 Impact on Equipment Behavior.

The only pieces of equipment located in the containment that were important for purposes of the Level 2 analysis were the cables to the safety/relief valve pilot valve solenoids, which were shown to be qualified to perform their intended function. The key plant damage state, "IAe1," which represents the limiting station blackout event, reached the maximum temperature of 320°F at 22 hours. The cables are rated to operate at 340°F.

2.3.5 Uncertainty and Sensitivity Analysis.

Section 4.6.2 of the submittal describes uncertainties of accident progression applicable to the BSEP in the following phases:

- Phase 1 - In-vessel thermal hydraulics
- Phase 2 - Uncertainties associated with the relocating core behavior and lower plenum phenomenology
- Phase 3 - Uncertainties associated with the ex-vessel phase following the ejection of the core debris from the reactor vessel.

The IPE team presented the source term results within an uncertainty range bounded by the 95th and the 5th percentiles, based on engineering judgment, which was reinforced by substantial experience and the IPE described herein. (See Tables 4.6.2-1a through 4.6.2-3, pages 4.6.200 through 4.6.205.)

2.4 Reducing Probability of Core Damage or Fission Product Release

2.4.1 Definition of Vulnerability.

The BSEP IPE submittal notes the following with respect to severe accident vulnerability (Section 3.4.2.1, pages 3.4.4 and 3.4.5):

> The team used the NUMARC Severe Accident Closure Guidelines (NUMARC 91-04) 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. The team reviewed the cutsets from the dominant accident sequences (above 1E-7/yr) and any associated component importance measures to select candidate items for comparison with the established criteria.

The NUMARC guidance is not quantitative and therefore is not measurable. By comparing the important issues identified in the IPE analysis (which are not listed in the submittal) and the list of plant modifications that BSEP had committed to make, the team discovered the following (page 3.4.5):

> The BSEP had already made commitments to install several modifications which would have an important beneficial effect on calculated core damage frequency, and that most of the IPE issues were already well along the way towards full resolution.

These modifications included hardened wetwell vents, a fifth diesel generator with dedicated switchyard batteries, and a remotely operated cross-tie. Between the time the CP&L IPE report was issued in August 1992 and a response was made to the NRC's RAI in February 1995, the hardened vent modifications for both units had been made. [1, 2] However, also during that same time period, the utility decided not to add a fifth diesel generator or the switchyard batteries. The reasons for this decision according to the submittal were that (1) the ability to cope with the loss of DC power had improved significantly because of added provisions for DC load shedding and (2) because in many cases DC power was available for 4 or 5 hours, significantly increasing the availability of depressurization.

2.4.2 Plant Improvements.

In response to CPI Program Recommendations, the IPE team made recommendations for plant improvements that are discussed in the next section.

2.5 Responses to CPI Program Recommendations

After considering the Containment Performance Improvement (CPI) Program recommendations, the IPE team concluded that, because of the BSEP design and the modifications that were in progress, no further actions were necessary to resolve the Mark I containment issues. As noted above, the hardened vent modifications were completed for both units; the utility decided not to add a fifth diesel generator or switchyard batteries.

The IPE did not take credit for the modifications to the vents. It did take credit for the BSEP adoption of Revision 4 of the Emergency Operation Procedures (EOPs) and operator training.

The IPE team concluded that supplying diesel-driven firewater to the containment sprays was of negligible benefit (Section 3.4.4, page 3.4.12):

> Marginal benefit in fission product scrubbing from the reduced flow that firewater system could provide, and the low shutoff head, nominally 120 psig, would preclude firewater injection into the containment or vessel during high pressure core damage sequences or in repressurization sequences.

It appears that the CP&L responses to the CPI Program recommendations are complete.

2.6 IPE Insights, Improvements and Commitments

The following insights reportedly were gained in performing the BSEP IPE:

- Unlike many other Mark I containments, the BSEP containment was considered unlikely to fail as a result of drywell liner melt-though, primarily because the liner was backed by and anchored into reinforced concrete. The unlikelihood of liner melt-though lowers the conditional probability of early containment failure.

- The contribution to containment failure by combustion processes was negligible because the BSEP containment was nitrogen-inerted. Hydrogen burns would structurally damage the reactor building, thus diminishing its effectiveness for fission product scrubbing.

- Apart from containment failure by pressurization at vessel breach, a leak that occurred before a break would prevent further pressurization and catastrophic containment failure.

- Hardened wetwell vent modification would significantly reduce the risk of severe accidents at BSEP.

- Accident sequences would lead to significant late containment failures because of the presence of a dry cavity and the absence of containment heat removal systems.

3. Contractor Observations and Conclusions

As discussed in Section 2 of this report, the IPE submittal contains a large amount of back-end information, which contributes to the resolution of severe accident vulnerability issues at BSEP. The submittal is well presented and describes what drives the IPE results. CP&L has identified plant improvements that could reduce severe accidents risks at BSEP. The utility has completed hardened vent modifications for both Units 1 and 2 of the Brunswick plant.

However, the following two issues are outstanding at the time of completing this TER. The NRC is in the process of their resolution with the utility.

- The possibility of high temperature failure of the cables to the safety/relief valve pilot valve solenoids which were the only pieces of equipment located in the containment that were important for the purposes of the Level 2 analysis.

- The reasons for the low likelihood of the drywell liner meltthrough leading to early containment failures.

4. References

1. Carolina Power & Light Company, "Brunswick Nuclear Plant Units 1 and 2 Individual Plant Examination Report," August 1992.

2. Carolina Power & Light Company, "Responses to Request for Additional Information Regarding the Review of IPE," February 1995.

3. D. A. Wesley, et. al., "Containment Overpressure Capacity for the Brunswick Nuclear Power Plant," EQE Engineering Consultants, December 1991.

Appendix
IPE Evaluation and Data Summary Sheet

BWR Back-End Facts

Plant Name

Brunswick

Containment Type

Mark I

Unique Containment Features

Containment liner is backed by a reinforced concrete wall.

Unique Vessel Features

None found

Number of Key Plant Damage States

5

Ultimate Containment Failure Pressure

131 psig (median value) (10% failure probability value of 105 psig was used for calculations)

Additional Radionuclide Transport and Retention Structures

Reactor building retention is credited.

Conditional Probability That The Containment Is Not Isolated

.0041

Important Insights, Including Unique Safety Features

Listed under Unique Containment Features

Implemented Plant Improvements

None

C-Matrix

KPDS	CDF (per rx year)	Early	Venting	Late	Intact
IAe1	1.8E-5	0.12		0.88	
IAd3	2.2E-7	0.01	0.99		
IAe3	4.7E-7	0.012		0.494	0.494
EAe1	7.7E-8	1.0			
YBe3	3.8E-8	1.0			