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SCIENTECH, INC.

CORPORATE HEADQUARTERS

1690 INTERNATIONAL WAY

IDAHO FALLS, IDAHO 83402

BEAVER VALLEY UNIT 1
TECHNICAL EVALUATION REPORT ON THE INDIVIDUAL PLANT EXAMINATION BACK-END SUBMITTAL

H. A. Wagage
J. F. Meyer

Prepared for the U.S. Nuclear Regulatory Commission
Under Contract NRC-05-91-068-32
August 1995

SCIENTECH, Inc.
11140 Rockville Pike, Suite 500
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E. EXECUTIVE SUMMARY

This technical evaluation report (TER) documents the results of the SCIENTECH submittal-only review of the back-end portion of the Beaver Valley Unit 1 (BV1) Individual Plant Examination (IPE) submittal.

E.1 Plant Characterization

Similar to the Surry nuclear power plant, the Beaver Valley Unit 1 plant is a pressurized water reactor (PWR) with a 3-loop nuclear steam supply system, designed by Westinghouse and engineered and constructed by Stone and Webster. Both plants have steel-lined, reinforced-concrete, subatmospheric containments. The major difference between the two plants is that Surry Unit 1 is rated for 775 MWe, and BV1 is rated for 833 MWe. In addition, BV1 is similar to Beaver Valley Unit 2 (BV2). The NRC staff completed its safety evaluation report on the BV2 IPE in May 1993. The BV1 containment cavity is not conducive to flooding.

E.2 Licensee IPE Process

The Duquesne Light Company (DLC) performed the IPE with support from Pickard, Lowe and Garrick (PLG), Inc., and from the Stone and Webster (S&W) Engineering Corporation. DLC reviewed the IPE in-house, in addition to and independent of the internal reviews conducted by PLG and S&W. The IPE team worked on site and participated in plant walk-throughs and inspections.

During the back-end analysis, the IPE team linked the Level 2 CET with the Level 1 event trees, quantifying all of the accident sequences from initiator to release category. The team used this quantification to analyze support systems and intersystem dependencies as well as to achieve the proper interface between the front-end and back-end analyses. Using this approach appears to have ensured that the support state conditions were properly accounted for throughout the front-end and back-end trees.

To quantify accident sequences, the BV1 IPE team used the computer code, RISKMAN, and the Surry NUREG/CR-4551 split fractions of top events and radionuclide release terms extensively. Although no plant-specific MAAP analysis was performed, the IPE team did use the results reported from the Beaver Valley Unit 2 (BV2) IPE, which were obtained using MAAP, Version 14. The IPE team used the results of some of the MAAP sensitivity analyses performed for BV2. Because similar designs lead to similar MAAP input values, the team's application of the Unit 2 results appears to have been reasonable. The IPE team used a mission time of 24 hours for the back-end analysis.

E.3 Back End Analysis

For Beaver Valley Unit 1, the IPE team calculated the total core damage frequency (CDF) from internal initiators to be 2.13E-4 per reactor year. One of the important groups of sequences that led to this CDF was loss of offsite power (23.9 percent) followed by loss of

emergency AC power train (19.3 percent). The next three contributors in descending order of magnitude were partial loss of main feedwater (12.3 percent), total loss of river water (11.2 percent), and non-isolable small LOCA (5.6 percent).

The containment event tree (CET) used in the Beaver Valley 1 plant IPE was similar to the accident progression event tree used in the NUREG-1150 analysis of the Surry nuclear power plant. The IPE team reviewed each of the 71 top events identified at the Surry plant for their applicability to the BV1 CET. After eliminating the events already contained in the BV1 plant damage states (PDSs) and combining several top events into single events, the IPE team identified 25 top events as applicable to the BV1 CET.

The BV1 CET was comparatively more detailed than the CETs used for many other IPEs. It integrated the systemic with the phenomenological aspects of severe accident progression. The IPE team did not take human actions into account in the CET for one of the following reasons: either procedural guidance did not exist, or because human actions were explicitly modeled in the front-end analysis, or because systems that might require operator actions (e.g., fan coolers) were not considered. Through the CET top events, the BV1 IPE team was able to directly address phenomenological issues concerning hydrogen burn, direct containment heating, steam explosions, molten core concrete interactions, and steam/noncondensible gas pressurization.

Members of the IPE team did not perform a plant-specific structural analysis of the BV1 containment. Instead, they compared and found that the design of the BV1 containment building was similar to that of the Surry 1 containment. In conducting the BV1 IPE, the team decided to use the Surry 1 containment failure distribution.

The BV1 IPE team used a definition of early containment failure that was more conservative than the one used in the NUREG-1150 study. For BV1, early containment failure was defined as occurring before or within 4 hours of vessel breach. In the NUREG-1150 study, it was defined as occurring before or within a few minutes of vessel breach. The IPE team defined large radionuclide releases as those with a source term equal to or greater than the PWR-4 source term of WASH-1400 (releases of 9-percent iodine and 4-percent cesium). The total frequency for large, early containment failures or small bypasses for BV1 was 1.06E-5 per reactor year.

The results of the back-end analyses at BV1 showed the following contributors to containment failure, given as a percentage of total CDF: early failure, 6.5; late failure, 43.4; containment bypass, 4.5; containment isolation failure, 16.3; and intact containment, 29.3. A modeling assumption that had a major impact on these results was that there was no in-vessel recovery after the initiation of core damage. By comparison, the Surry plant had a 46.7 percent in-vessel recovery. Two of the containment characteristics that drove these contributors to containment failure were 1) the relatively small volume of the containment, making DCH and hydrogen burns more important, and 2) the high probability that the reactor cavity would be dry.

The 16.3-percent CDF that the team calculated for containment isolation failures is relatively high, e.g., 0.2 percent for the North Anna IPE and 1.0 percent for the Zion NUREG-1150 study. However, DLC has defined containment integrity conservatively: Containment isolation failure size was assumed to be less than 3 inches in diameter. Because the BV1 containment is subatmospheric, larger openings were not expected to exist. This definition appears to be more conservative than that used in many IPEs where the diameter of the opening for containment isolation was assumed to be greater than 2 inches. At BV1 isolation failures were not excluded based on small size and openings up to 1/8 inch in diameter were investigated. The isolation failures found at BV1 were 2 inches and 1 inch in diameter. During the IPES at other plants, an opening of diameter 2 inches or less did not cause containment isolation failure, whereas, at BV1, it did.

The small containment isolation failure plant damage states contributed so much to the CDF because the majority of the failures (96.2% of isolation failures or 15.5% of the total CDF, based on the saved sequence database) were due to emergency switchgear ventilation failure, which resulted in the guaranteed failure of all emergency power and, consequently, in containment isolation. The normally open reactor coolant pump (RCP) seal return line requires AC power to close. Failure to isolate the RCP seal return line was modeled as a failure of containment isolation.

The IPE team presented radionuclide releases in terms of 20 release categories and four major release category groups (RCGs), I through IV. The descriptions and percentile contributions to the CDF from RCGs were as follows: large, early containment failures and bypasses, 5.0 (1.06E-5 per year); small, early containment failures and bypasses, 22.3; late containment failures, 43.4; and long-term containment releases (intact containment), 29.3.

In most of the accident sequences, the BV1 containment cavity stayed dry. Even in those sequences where the cavity was flooded, the IPE team argued that the containment sprays were operational, making it less important that the aerosol was removed by the water pool overlying the melt. Therefore, in characterizing the BV1 source terms, the team members assumed that the cavity stayed dry during all of the accident sequences. (The presence of water in the cavity, however, was considered in calculating core-concrete interactions.)

BV1 has a smaller subatmospheric containment than do other PWRs with large, dry containments. This makes the issue of hydrogen combustion more important for BV1. For example, 40-percent oxidation of the core zircalloy would raise the molar concentration of hydrogen in the dry air of the BV1 containment on the order of 10 percent, while 100-percent oxidation of zircalloy would raise the concentration on the order of 22 percent. In analyzing hydrogen combustion at BV1, the IPE team used assumptions more conservative than those used during the NUREG-1150 study of the Surry plant (e.g., members of the IPE team used a detonation limit of 12 percent compared with the 14-percent limit used in the Surry analysis). The IPE team assumed that hydrogen detonations would always fail the containment.

Containment failures at vessel breach caused by high-pressure melt ejection (HPME) were due in part to hydrogen combustion. The pressure rise in vessel breach represented the combined effects of blowdown, hydrogen burning, DCH, and steam explosion. The IPE

team assumed that large quantities of hydrogen would be available for combustion before and after vessel breach. If this hydrogen had not burned before or at vessel breach and the containment atmosphere was not inerted by steam, there was a high probability that a burn would occur in the "early" time frame, within 4 hours of vessel breach. Based on the lower flammability limit of 4 percent hydrogen the in air, the team assumed a probability of 0.965 that an "early" hydrogen burn would occur if containment sprays were operating and no HPME occurred at vessel breach. If debris was cooled and the containment sprays were operating, no significant source of additional hydrogen would be present in the early time frame, and burns were assumed to be unlikely. If an early hydrogen burn did occur, the containment failure was assumed to be a certainty.

The IPE team assumed that, if the debris was not being cooled, there was a 50-percent chance that there would be a hydrogen burn in the "late" time frame (more than 4 hours after vessel breach) if the containment was inerted, i.e., sprays were not operating. If the containment was not inerted (i.e., electrical power was available and sprays were operating) and debris was not cooled, it was assumed that late hydrogen burns would occur. If the debris was being cooled and the containment sprays were operating, no significant source of hydrogen would be present, and therefore late hydrogen burns would be unlikely.

Percentile contributions of hydrogen burn/deflagration-to-detonation transition to the various release classes were reported as follows (page 58 of Dusquesne's response to the RAI): [3]

- 0.2% of release category group (RCG) I - large, early containment failures and bypasses
- 0.0% of RCG II - small, early containment failures and bypasses
- 11.3% of RCG III - late containment failures.

It appears that the IPE team gave adequate consideration to the CPI Program recommendations on issues related to hydrogen combustion. The IPE team addressed hydrogen pocketing issues during the containment walkdown.

Based on the MAAP analysis performed in the BV2 IPE, the BV1 IPE team concluded that induced steam generator tube rupture (ISGTR) was not a major concern for BV1. Team members predicted that hot leg or surge line failure would occur during high-RCS pressure sequences prior to ISGTR.

Early overpressurization of the containment was a major contributor to the early, large containment failure category at BV1. (For comparison purposes the major contributor to such failures at the Surry plant was containment bypass). According to the submittal, direct containment heating, which is caused by high-pressure melt ejection, strongly influenced early overpressurization failure of the containment. A major reason for the strong influence from HPME was a modeling assumption that did not allow for operator depressurization, thus increasing the likelihood of vessel breach at high pressure.

E.4 Containment Performance Improvements (CPIs)

In response to Containment Performance Improvement Program recommendations, the BV1 IPE team did the following:

- Visually inspected the containment geometry and openness and the location of potential ignition sources for combustible gases during the containment walkdown performed to gather information for the back-end analysis
- Evaluated and found that the containment penetration seals were not vulnerable to thermal attack from hot combustion gases
- Addressed important issues regarding hydrogen generation and hydrogen combustion/detonation
- Analyzed the vulnerability of containment performance to hydrogen combustion using BV1 CET top events.

E.5 Vulnerabilities and Plant Improvements

The IPE team identified a back-end improvement which was an enhancement to update procedures to depressurize primary and secondary systems. The existing EOPs at BV1 were not explicit for sequences where high head safety injection was unavailable. This improvement is to be considered in the BV1 Accident Management Program.

The IPE team identified two back-end plant vulnerabilities based on major contributors to large, early radionuclide releases. The first vulnerability involved phenomena leading to containment overpressurization during a core-melt sequence that included the RCS blowdown, early hydrogen burns, and DCH. In this regard, the IPE team identified several actions that could be taken to lower containment overpressurization: lower the RCS pressure before vessel breach, flood the reactor cavity, and establish debris cooling after vessel breach.

The second vulnerability that the team identified was containment bypass from interfacing system loss of coolant accidents (LOCAs) and steam generator tube ruptures (SGTRs). Interfacing system LOCAs contributed 10.7 percent to the large, early releases compared with 77 percent for the Surry plant. This relatively low contribution resulted from there being only one interfacing system, the LHSI, located outside the containment at BV1. (The RHR system is located inside the containment.) The SGTRs contributed 3.2 percent to the large, early releases at BV1 compared with 10 percent for the Surry plant. Thus, the combined contribution of interfacing system LOCAs and SGTRs to large, early releases at BV1 was 13.9 percent compared with 87 percent for the Surry plant. Changes to plant procedures and training were being implemented to enhance the operator response to such sequences. For "LOCA outside the containment," the IPE team identified the importance of improving guidance to the operators on the key valve to close.

E.6 Observations

SCIENTECH found DLC treatment of radiological releases to be satisfactory and consistent with Generic Letter 88-20. SCIENTECH noted the following strengths in the BV1 IPE back-end analysis:

- In order to treat properly the dependencies between the front-end safety systems that are needed to prevent damage to the core, to the containment system, and to the support systems that tie both together, the IPE team included all the active containment systems (e.g., quench and recirculation sprays, containment isolation) in the front-end trees.

- Containment event tree development and quantification in the BV1 IPE is very thorough, well presented, and in accordance with the level of detail requested in the GL 88-20 and NUREG-1335.

- It appears that the IPE team identified all relevant potential containment failure modes. All applicable containment failure modes that appear in Table 2-2 of NUREG-1335 were considered.

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 Beaver Valley Unit 1 (BV1) Individual Plant Examination (IPE) submittal. [1, 3] This TER was prepared to comply 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

In October 1994 SCIENTECH delivered a draft TER for the back-end portion of the BV1 IPE submittal to the NRC. Based in part on this draft submittal, the NRC staff submitted a Request for Additional Information (RAI) to Duquesne Light Company. Duquesne Light Company responded to the RAI in a document dated March 10, 1995. 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 BV1 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

Similar to the Surry nuclear power plant, the Beaver Valley Unit 1 plant is a pressurized water reactor with a 3-loop nuclear steam supply system, designed by Westinghouse and engineered and constructed by S&W. Both plants have steel-lined, reinforced-concrete, subatmospheric containments. The major difference between the two plants is that Surry Unit 1 is rated for 775 MWe, and BV1 is rated for 833 MWe. The BV1 containment data and design description are provided in detail in Section 4.1 of the IPE submittal. The principal design parameters and characteristics of the BV1 containment are summarized and compared with their Surry Unit 1 counterparts in Table 4.1-1 of the submittal.

The BV1 containment building is a steel-lined concrete shell with a vertical cylinder, a hemispherical dome, and a flat base. The flat base is 10 feet thick and supports the cylinder and the cylindrical dome. The cylinder is 4 feet, 6 inches thick and 122 feet, 1 inch high, and is covered with a 3/8-inch-thick steel liner. The 2-foot, 6-inch-thick dome has a 63-foot radius and is covered with a 1/2-inch-thick steel liner.

Table 1 compares the key design features of the BV1 plant containment systems with those of the North Anna and Surry plants. This comparison indicates that all three containment systems are similar. (The Containment Capacity Measure for BV1 is smaller than it is for North Anna and Surry. But the difference in values (1.75 for BV1, and 1.85 each for the other two) does not indicate that the BV1 containment is less robust than the North Anna or Surry containments.) In addition, the BV1 containment includes the following important features:

- The reactor vessel is surrounded by an annular neutron shield tank, surrounded by the concrete primary shield wall. A gap of 2-3/4 inches between the reactor vessel and shield tank permits water drainage from the refueling tank to the reactor cavity. The water in the containment proper would have to rise to a level of 14 feet, 7 inches, before it would overflow to the cavity through the instrument tunnel. Conversely, water that entered into the cavity from the refueling tank would be unavailable for recirculation until the cavity and instrument tunnel were filled to a level of 16 feet, 7 inches, above the cavity floor.

- As noted above, the BV1 reactor vessel is supported by the shield tank, which, in turn, is supported by a steel support skirt. An annular ring of steel-clad lead shielding surrounds the skirt. During a severe accident involving a release of molten core into the cavity, heating of the cavity atmosphere could melt the lead shield and add up to 100,000 lbm of lead to the melt pool on the cavity floor.

- The upper floor of the containment has many openings that permit containment spray water to drain to the bottom floor of the containment and to the sump. The opening to the sump is protected by large vertical bars, a coarse mesh screen, and a fine mesh screen.

- The BV1 containment fan coolers do not perform a safety function. They trip on a safety signal and therefore are unavailable in response to most accidents.

- The configurations of structures and of equipment inside the containment were found to be conducive to good air circulation. The steam generator, pressurizer cubicles, and most compartments within the containment are open at their tops to the general containment atmosphere.

Table 1: Summary of Key Plant and Containment Design Features for the Beaver Valley Plant

Characteristic	North Anna	Surry	Beaver Valley 1
Thermal Power, MWt	2,893	2,441	2,652
RCS Water Volume, m^3	282	260.6	259
Containment Free Volume, m^3 (ft^3)	51,680 (1.83E6)	49,000 (1.73E6)	49,800 (1.76E6)
Mass of Fuel, kg	82,160	79,637	82,179
Mass of zircalloy, kg	17,108	16,463	17,338
Mean Failure Pressure, psia	143	141	141
Containment Capacity Measure*	1.85	1.85	1.75

* Containment Capacity Measure = $\dfrac{[\text{Containment Free Volume, ft}^3] \times [\text{Mean Failure Pressure, psia}] \times 10}{[\text{Mass of Fuel, kg}] \times [\text{Mass of zircalloy, kg}]}$

2. TECHNICAL REVIEW

In conducting the "submittal only" review, SCIENTECH compared the Beaver Valley Unit 1 IPE submittal with the requirements of Generic Letter (GL) 88-20 and its supplements, using guidance provided in NUREG-1335. In presenting our review findings we used the structure of Task Order Subtask 1 and followed the key requirements of the GL and its supplements. Inconsistencies between the BV1 IPE and other probabilistic risk assessment (PRA) studies in terms of the methodology used and the results obtained are noted, and the BV1 IPE strengths and weaknesses are identified.

2.1 Licensee IPE Process

2.1.1 Completeness and Methodology.

The BV1 IPE submittal contains a substantial amount of information in accordance with the requirements of GL 88-20, its supplements, and NUREG-1335. The submittal appears to be complete and to provide the level of detail requested in NUREG-1335.

The methodology used to conduct the IPE at BV1 is described clearly in the submittal and the approach taken is consistent with the basic tenets of GL 88-20, Appendix 1. This approach and the basic assumptions underlying it are described clearly. The important plant information and data are well documented and the key IPE results and findings are well presented.

The containment event tree (CET) used in the Beaver Valley 1 plant IPE was similar to the accident progression event tree used in the NUREG-1150 analysis of the Surry nuclear power plant. The IPE team reviewed each of the 71 top events identified at the Surry plant for their applicability to the BV1 CET. After eliminating the events already contained in the BV1 plant damage states (PDSs) and combining several top events into single events, the IPE team identified 25 top events as applicable to the BV1 CET (excluding the entry state top event).

The entry state to the CET was either a PDS or a Level 1 event tree. The Level 1 core damage frequency (CDF) sequences were binned into several PDSs with similar challenges presented to the containment. After considering plant conditions, systems, and features that can have a significant impact on the potential course of an accident, the IPE team identified a total of 640 possible PDSs. This number was later reduced to 143 by eliminating some combinations of some PDS characteristics. Because of this large number, the IPE team found that it was more convenient to quantify the Level 2 CET by physically linking it to the Level 1 event trees and quantifying the entire accident sequence frequencies from initiator to release category.

This quantification was applied to analyze support systems and intersystem dependencies as well as to achieve the proper interface between the front-end and back-end analyses. Using this approach appears to have ensured that the support state conditions were properly accounted for throughout the front-end and back-end trees. However, it is not easy to trace

and review the results obtained from a very large integrated risk model involving a very large number of accident sequences.

To quantify accident sequences, the BV1 IPE team used the computer code, RISKMAN, and the NUREG/CR-4551 split fractions of top events and radionuclide release terms extensively. Although no plant-specific MAAP analysis was performed, the IPE team did use the results reported from the Beaver Valley Unit 2 (BV2) IPE, which were obtained using MAAP, Version 14. Because similar designs lead to similar MAAP input values, the team's application of the Unit 2 results appears to have been reasonable.

The BV1 IPE team defined early containment failure as that occurring before or within 4 hours of vessel breach. This definition is different from the one used in the NUREG-1150 study in which the early containment failure of a pressurized water reactor (PWR) was defined as occurring before or within a few minutes of vessel breach. Therefore, some containment failures that were considered early during the BV1 IPE would be binned as late under the NUREG-1150 definition. The BV1 IPE team used a 24-hour mission time in its analysis.

2.1.2 Multi-unit effects and As-built, As-operated Status.

Beaver Valley Units 1 and 2 have containment buildings that are very similar in design and function. Therefore, in developing the Unit 1 Level 2 back-end analysis model the IPE team used the same logic and some of the split fraction values of the back-end model used for Unit 2.

The Beaver Valley Unit 1 IPE team was located on site and participated in plant walk-throughs and inspections. The team did a 1-day plant familiarization walkthrough in November 1988 and, for purposes of Unit 1 back-end analysis, a half-day containment walk-through in September 1989. During the back-end containment walkthrough, the IPE team members made a general inspection of the geometry and "openness" of the containment and of the location of potential ignition sources for combustible gases. They focused specifically on the following:

- Configuration of the reactor cavity and instrument tunnel
- Pathways to and location and configuration of the containment sump
- Equipment hatches and personnel air locks
- Containment penetrations (Section 4.1.2.1, pages 4.1-2 and 4.1-3).

Before the walkthrough, the team compared the BV1 with the BV2 and Surry Unit 1 containment designs to identify any design conditions that could cause the BV1 containment to behave differently if subjected to high pressure and temperature. In addition, the Duquesne Light Company independent review team verified that the models reflect actual plant design and operation. It appears that the BV1 containment-specific features were modeled.

2.1.3 Licensee Participation and Peer Review of IPE.

In 1988, DLC initiated plans to develop PRAs on BV1 and BV2. The nuclear engineering department was responsible for developing the technical capability to complete the PRAs with support from Pickard, Lowe and Garrick (PLG), Inc., and from the Stone and Webster (S&W) Engineering Corporation. To gain in-house knowledge and maximize the benefit from performing PRAs, the DLC did the following (Section 5.1, page 5.1-1 of the submittal):

- Became involved in all aspects of the BV2 PRA. With support from PLG and S&W, DLC developed the technical capability to perform PRAs, and updated the BV2 PRA to use plant-specific data and reflect plant changes made in 1990 and 1991.

- Led the BV1 PRA team, based on knowledge gained in performing the BV2 PRA.

- Performed reviews of both PRAs, in addition to and independent of the internal reviews conducted by PLG and S&W.

- Understood both PRAs in sufficient detail to be able to present and use their results with minimal support from PLG and S&W.

In support of the BV1 PRA, the Engineering Analysis Assurance (EAA) group provided interface with other DLC departments to ensure that assumptions were correct about plant design and capabilities, success criteria, and the implementation of emergency operating procedures. DLC involvement included the following:

- Three DLC engineers participated in the BV2 containment walkthrough and later the BV1 walkthough in support of the Level 2 "Back-End Analysis."

- DLC EAA engineers developed the MAAP input parameter file and ran cases with support from S&W.

- DLC training and operations personnel reviewed the important operator actions and provided input to the quantification and accuracy of human actions.

- DLC prepared the BV1 IPE submittal.

The DLC independent review team included personnel selected from certain departments in order to provide specific knowledge of BV1 plant design, system configuration, and operating procedures. The IPE team responded to comments and recommendations from PRA team members, supporting departmental personnel, and DLC independent review team members on a continual basis. The DLC independent review verified that the model reflected actual plant design and operation. The review also enhanced awareness and knowledge of PRA throughout the DLC organization.

It appears that DLC involvement in the IPE was significant and that the IPE received adequate internal and external peer reviews.

2.2 Containment Analysis/Characterization

2.2.1 Front-end Back-end Dependencies.

As described in Section 2.1.1 of this report, the IPE team developed PDSs which were used for purposes of presentation and understanding only, and not for the CET analysis. In selecting the entry conditions to the CET analysis, the team chose the Level 1 trees. The submittal notes that this linking of Level 2 CET directly to the Level 1 trees (Section 4.5, page 4.5-1):

> greatly facilitates the treatment of dependencies between Level 1 and Level 2 events, thus satisfying the 'cross-checking' concerns raised by the NRC in NUREG-1335.

In order to treat properly the dependencies between the front-end safety systems that are needed to prevent damage to the core, to the containment system, and to the support systems that tie both together, the IPE team included all the active containment systems (e.g., quench and recirculation sprays, containment isolation) in the front-end trees. It appears that the IPE team's treatment of front-end back-end interface dependencies was complete and in accordance with the level of detail requested in NUREG-1335.

2.2.2 Containment Event Tree Development.

The BV1 IPE model consisted of a directly linked set of system event trees (front-line and support) and containment event trees: each accident sequence was treated from the initiating event to the release of radionuclides. The probabilistic quantification of severe accident progression was performed using the containment event tree approach. The CET was used to map out the possible containment conditions affecting the radionuclide releases associated with a given core damage sequence (or class). The BV1 IPE team used a detailed CET, which integrated systemic with phenomenological aspects of severe accident progression. The team did not take human actions into account in the CET for one of the following reasons: either because procedural guidance did not exist, or because human actions were explicitly modeled in the front-end analysis, or because systems that might require operator actions (e.g., fan coolers) were not considered, as described in Section 2.2.5 of this report.

The IPE team reviewed each of the 71 top events used in the NUREG-1150 study of Surry for their applicability to the BV1 CET. After eliminating events already contained in the BV1 PDSs and combining several top events into single events, the team identified the following 26 top events as applicable. These include the entry state top event (no. 1).

- 1. Dummy event or PDS (IE)
- 2. Core damage state at the end of Level 1 trees (SS)
- 3. Failure to arrest core damage and prevent vessel breach (CP)
- 4. Containment bypass prior to core damage (BY)
- 5. Large bypass prior to core damage (BL)
- 6. Induced PORV failure (LS)
- 7. RCP seal LOCA (SP)
- 8. Induced steam generator tube rupture (IS)

- 9. Induced RCS hot leg or surge line failure (IP)
- 10. RCS pressure at vessel breach (RP)
- 11. Containment failure prior to vessel breach (C1)
- 12. Large containment failure prior to vessel breach (L1)
- 13. In-vessel steam explosion that fails containment (AP)
- 14. High-pressure melt ejection (ME)
- 15. Containment failure at vessel breach (C2)
- 16. Large containment failure at vessel breach (L2)
- 17. Hydrogen burn within 4 hours of vessel breach (HE)
- 18. Containment failure due to early hydrogen burn (CE)
- 19. Large containment failure from early hydrogen burn (LE)
- 20. Failure to cool debris (DC)
- 21. Late burn of combustible gases (H3)
- 22. Late containment failure due to burn (C3)
- 23. Large, late containment failure due to hydrogen burn (L3)
- 24. Long-term overpressurization (C4)
- 25. Long-term overpressurization that causes large containment failure (L4)
- 26. Basemat penetration (BI)

Top Event 1 involved the entry state from the Level 1 trees. Top Events 2 and 3 involved degraded core recovery, where core cooling is recovered after the core is degraded or has melted, but before the vessel is breached. Degraded core recovery was limited mainly to core damage phases before significant core relocation to the bottom of the vessel occurred.

Top Events 2 through 12 involved the analysis of phenomena that occurred while the core was contained inside the vessel. Top Events 13 through 20 involved the analysis of phenomena that occurred immediately after vessel breach and lasted until debris quenching or dryout occurred. Top Events 21 through 26 involved the analysis of phenomena that occurred over the long term, after the quenching or the drying out of the debris. As seen in the above list, the BV1 CET included phenomenological questions as CET top events if they addressed one of the following issues (Section 4.5-1, page 4.5-2):

- Definition of a safe, stable state for the debris configuration either in-vessel or in the containment

- Dependencies of later top events in the CET, such as RCS pressure at vessel breach, high-pressure melt ejection, hydrogen burn, basemat melt-through

- Containment failure events and failure modes.

Phenomenological issues concerning hydrogen burn, direct containment heating, steam explosions, molten core concrete interactions, and steam/noncondensible gas pressurization were addressed directly by CET top events. Split fraction values for CET top events reflected finite probabilities of different paths and thus they included uncertainties. A detailed description of split fractions is given in Table 4.6.4, pages 4.6-32 through 4.6-41 of the submittal. (See Section 4.6.3 of the submittal for quantification.)

Containment event tree development and quantification in the BV1 IPE is very thorough, well presented, and in accordance with the level of details requested in the GL 88-20 and NUREG-1335.

2.2.3 Containment Failure Modes and Timing.

The IPE team compared the design of the BV1 containment building with that of the Surry 1 containment to determine if the containment failure distribution identified for Surry could be applied to BV1 with a high degree of confidence. A summary of this comparison is given in Table 4.1-1, pages 4.1-9 through 4.1-16 of the submittal, and details are given in Table 4.1-2, pages 4.1-17 through 4.1-42. The comparison focused on properties that would affect the relationships among pressure, temperature, and containment integrity. In particular, the following design attributes were analyzed:

- Containment geometry
- Material properties
- Rebar quantities and pattern
- Liner thickness
- Hatches and penetration configurations
- Calculated and actual pressure test data.

Based on this comparison, the IPE team concluded that, despite some minor differences, the BV1 and Surry containments were similar. Because the density of the rebar was somewhat higher in some areas of the BV1 than in the Surry 1 containment, the team conservatively applied the Surry 1 containment failure distribution in conducting the BV1 IPE. This application of the Surry failure distribution to BV1 appears to have been reasonable.

The mean containment failure pressure for Surry as found in the NUREG-1150 study is 126 psig.

The IPE team analyzed the following three failure mechanisms: containment penetration leakage, reactor pressure vessel support failure, and containment failure modes associated with anticipated transients without scram (ATWS) events. The team assessed the nonmetallic seals used for large containment penetrations of BV1 (personnel air lock, equipment hatch, and emergency air lock) and found that the seals were similar to those of the Surry plant. The submittal notes that the seal material used for BV1 is the EPDM compound. Based on the findings reported in NUREG-1037 [4], the team concluded that large containment penetrations would not be a significant source of containment leakage.

A shield tank supports the BV1 reactor vessel at the primary loop nozzles and transfers vertical loads down through a skirt to the containment mat. As the result of accident sequences involving vessel melt-through, the base of the skirt would temporarily contain debris and would be likely to fail in the event of thermal attack by debris. The team investigated and determined that containment breach would be unlikely as the result of the RPV support failure.

The IPE team considered severe accident sequences in which the size of the reactor coolant pressure boundary (RCPB) significantly exceeded the double-ended rupture of the largest coolant pipe that constitutes the emergency core cooling system (ECCS) design basis. For large, dry containments, the submittal notes, the containment pressure response to such events, referred to as "excessive LOCAs," was within the design basis. Therefore, the team focused on other containment failure modes. The main contributors to excessive LOCAs at BV1 were ATWS events involving the failure of primary system pressure relief. The team focused on the three weakest elements of the RCPB (control rod drive housing and bolts, residual heat removal gate valve, and reactor coolant pump casing and bolts) and found that such containment failures were unlikely. The submittal notes that, for BV1, the residual heat removal system was entirely within the containment so that failure of the gate valve disk would not lead to containment bypass.

It appears that the IPE team identified all relevant potential containment failure modes. All applicable containment failure modes that appear in Table 2-2 of NUREG-1335 were considered.

2.2.4 Containment Isolation Failure.

In the BV1 IPE, containment isolation (CI) was modeled in the front-end analysis. CI was the last top event of the front-line event trees for the transient/small LOCA, medium LOCA, and large LOCA. This top event questioned the failure to create and maintain an isolated containment following safety injection and CIA and CIB signals. The following containment penetrations were modeled (page 3.1-70 of the submittal):

- Containment major vents and drains, e.g., sump pump discharge
- Connections to RCS, e.g., RCP seal water return
- Connections to containment atmosphere, e.g., containment vacuum line.

This top event also modeled the operator actions to ensure that isolation valves remained closed after the CIA and CIB signals were reset. Manual isolation of the RCP seal return line during a loss of vital AC was also modeled in this top event.

Containment isolation failure size was assumed to be less than 3 inches in diameter. Because the BV1 containment is subatmospheric, larger openings were not expected to exist. This definition appears to be more conservative than that used in many IPEs where the diameter of the opening for containment isolation was assumed to be greater than 2 inches. At BV1 isolation failures were not excluded based on small size and openings up to 1/8 inches in diameter were investigated. The isolation failures found at BV1 were of 2 inches and 1 inch in diameter. At other plants, an opening of diameter 2 inches or less did not cause containment isolation failure, whereas, at BV1, it did.

Containment isolation failures contributed to 16.3 percent of the total BV1 CDF (or 3.48E-5 per year) (Table 1-3, page 1.4-7). This value is significantly higher than values resulting from other PRAs, e.g., 0.2 percent for the North Anna IPE and 1.0 percent for the Zion

NUREG-1150 study. The reason that the small containment isolation failure plant damage states contribute so much to the CDF is because the majority of the failures (96.2%, based on the saved sequence database) are due to the emergency switchgear ventilation failure (15.5% of the total CDF), which results in the guaranteed failure of all emergency power and consequently, containment isolation. The normally open reactor coolant pump (RCP) seal return line requires AC power to close. Failure to isolate the RCP seal return line was modeled as a failure of containment isolation.

2.2.5 System/Human Response.

The BV1 IPE back-end analysis did not include human interaction events for the reasons given below:

- Operator depressurization before core damage was addressed in the front-end analysis. Operator depressurization after core damage was outside the scope of existing emergency procedures and was left for consideration under accident management (No. 5, page 4.5-11).

- AC power recovery was not included in the CET because procedural guidance after core damage was unavailable (page 4.6-10; no. 45, page 4.5-16).

- Status of the quench sprays system and of the recirculation spray system was determined in the Level 1 model and not explicitly treated as a CET top event.

- Recirculation fan cooling system was not designed to operate under severe accident conditions and therefore its status was not included as a CET top event.

In those accident sequences where the loss of emergency switchgear ventilation led to a station blackout, the IPE team did not take credit for operator actions taken to manually isolate the containment building. However, the team did take credit for operator actions in the case of an SBO that resulted from the loss of the offsite power initiator and where both emergency AC power trains failed. (Section 4.8.2). For BV1, containment isolation failure was responsible for 72 percent of the sequences that made up Release Category Group II (small, early containment failures and bypasses). The submittal notes that containment isolation failures might be reduced if the operators were allowed to manually isolate the containment building following failures of the emergency switchgear ventilation.

2.2.6 Radionuclide Release Categories and Characterization.

The IPE team developed release categories, which are groups of CET end states that can be represented by similar source terms. The variations in source terms for accident sequences within one release category are smaller than the variations from one release category to another. The team considered the following elements and their states of condition in terms of how they might affect the definition of release categories for large, dry PWR containments (Table 4.7-1, pages 4.7-11 and 4.7-12):

- Containment bypass (e.g., yes, no)

- RCS pressure (e.g., high, moderate, low)

- Time of containment breach (e.g., preexisting (i.e., before core degradation), early, late)

- Size of containment breach (e.g., large, small)

- Location of containment breach (e.g., through auxiliary building, outside auxiliary building, basemat melt-through, induced containment bypass)

- Spray system (e.g., available through the accident, partially available (i.e., up to the time of containment failure), not available)

- Reactor cavity (e.g., wet, dry)

Separate release categories were assigned for the BV1 containment bypass. In considering the effect of the above on BV1, two high-level simplifications were made (pages 4.7-1 and 4.7-2):

- 1. The issue of the presence of water in the cavity was eliminated in the definition of release categories based on the following:

  Any or all of the following conditions could add water to the cavity:

  - Operation of quench spray pumps before vessel failure

  - Presence of water in the vessel lower head, or accumulator discharge (if the reactor pressure was sufficiently high to prevent discharge) after vessel failure

  - Failure of the shield tank after vessel failure.

  However, the amount of water that would be added to the cavity in any of these cases would not be sufficient to keep the core debris covered for more than a few hours. If the low head safety injection (LHSI) pumps or the two RSS pumps lined up for vessel injection continued to run after vessel failure, water would be present in the cavity. The submittal states that this situation would occur after a successful quench and recirculation spray operation and lineup for vessel injection, but with ECCS failure. Because the spray operation would scrub the aerosols generated by core-concrete interaction, it was less important for the source term whether the cavity was "wet" or "dry." Thus, the issue of the presence of water in the cavity was eliminated.

- 2. Based on ANS study results, [2] the IPE team decided to combine "preexisting" and "early" containment failures.

In addition, the IPE team made several other simplifications as described and presented in detail in Section 4.7.1.2. The simplifications of release categories appear reasonable. A total of 17 release categories were developed, excluding the three release categories for bypass sequences. The source terms for these categories were determined based on the following (Section 4.7.2.1, page 4.7-5):

- Review of Surry analyses
- Adaptation of existing Surry source terms to appropriate BV1 release categories
- Development of the source terms for BV1 release categories for which Surry analysis did not exist or for which there were several analysis with differing results.

In all cases (except intact containment cases), the team assumed a 100-percent noble gas release. For a major fraction of release categories, the source terms were based on the results of several existing analyses (BMI-2104, BMI-2139, NUREG/CR-5082, NUREG-0958 Draft, NUREG/CR-4828, NUREG-0956, and an ANS report). For other release categories, the source terms were determined from a limited number of analyses (primarily draft NUREG-0956) and from source terms for similar release categories using correlation factors to account for the effects of the following: operation of sprays for early containment failures, containment failure timing (early or late), and minimum limit of iodine release. The IPE team used a significant amount of available data for source term quantification. This application of available results to BV1 appears to have been reasonable and consistent with the requirements of GL 88-20.

Table 4.7-7, page 4.7-18, lists the characteristics of the 21 BV1 release categories, and Table 4.7-10, page 4.7-32, lists the release fractions of 20 release categories, excluding Release Category 21, which involved sequences with containment intact, in terms of I (CsI), Cs (CsOH), Te, Ba/Sr, and La. Release energies and release timing (start time and duration) for these release categories are given in Sections 4.7.2.2 and 4.7.2.3 of the submittal, respectively.

NUREG-1335 sets out reporting guidelines for systemic sequences, which stipulate that "all systemic sequences within the upper 95% of the total containment failure frequency" be reported to the NRC. However, arguing that late containment failures have much less potential than early ones for posing a threat to public health, the IPE team decided not to report late containment failures and instead reported the top 100 sequences for each of the following:

- Release Category Group I : Large, early containment failures or large bypasses with source terms equal to or greater than the PWR-4 source term of WASH-1400 (releases of 9% iodine and 4% cesium)
- Release Category Group II: Small, early containment failures or small bypasses with source terms less than the PWR-4 source term of WASH-1400.

2.3 Accident Progression and Containment Performance Analysis.

2.3.1 Severe Accident Progression.

Section 4.6.2 of the submittal describes the analyses of accident progressions that were performed as part of the BV2 PRA and that are applicable to the BV1 IPE. The probabilities of induced steam generator tube rupture (ISGTR) or induced hot leg failure before vessel breach from fast SBO scenarios with reduced coolant pump seal leakage were assessed at BV2 using temperature and pressure data obtained using the MAAP computer code (Version 14). Being a plant of similar design, BV1 has a similar MAAP input file. A major difference, however, is that the BV2 steam generator inventory was 28% larger than the BV1. This larger inventory extended the time to steam generator dryout by about 0.6 hours (at BV1, dryout occurred 0.6 hours earlier) but was not likely to affect the temperature levels much. (The BV1 recovery times would be less than those for BV2.)

The BV2 IPE team performed several base case and sensitivity analyses of the occurrence of potential hot leg or steam generator creep rupture. The first base case calculated for fast SBO with the reactor scrammed and isolated, the RCP stopped, no feedwater flow available (i.e., the turbine driven pumps failed), steam generator pressure maintained at about 1,060 psig, and an RCP seal leak rate of 21 gpm/pump equivalent; i.e., the low end of the expected leak rate based on NUREG-1150 expert opinion (page 4.6-2). MAAP analysis showed that, at the above leak rate of 21 gpm/pump equivalent, the RCS pressure remained at the PORV set point of about 2350 psia at the vessel breach; for higher seal leak rates, the vessel pressure dropped substantially. The IPE team ran two MAAP cases: one with core blockage ON, and one with core blockage OFF. In both cases, it was predicted that steam generator dryout would occur in about 1.8 hours, and core uncovery in about 2.2 hours. In the Blockage OFF case, the team predicted a slightly delayed time for vessel breach (3.8 hours, versus 3.7 hours for the Blockage ON case) and somewhat higher peak pressure boundary temperatures: peak hot leg temperature (1,900°F, versus 1,500°F in the ON case) and peak steam generator tube temperature (800°F, versus 770°F in the ON case). Because it posed a higher threat to steam generator tube integrity, the Blockage OFF case was chosen as the first base case.

The second base case consisted of a slow SBO, similar to the SBO in the first base case, except that turbine-driven feedwater was available initially but failed later as a result, for example, of failures in running the turbine or failures resulting from DC power loss caused by battery drain. Using MAAP results for the two base cases and sensitivity analysis cases, the team calculated the possible creep rupture of the following RCS pressure boundary components exposed to high temperatures:

- Low-alloy steel reactor vessel hot leg nozzle safe ends
- 316 stainless-steel hot legs and surge lines
- Inconel 600 steam generator tubes.

Using the results of the above analyses, the BV1 IPE team concluded that hot leg failures were likely to occur before vessel melt-through and that ISGTR was unlikely.

Two methods can be used to achieve in-vessel recovery: flooding the reactor cavity, and depressurizing the primary system to allow LHSI. Flooding the cavity to a level sufficient to cool the bottom head before vessel breach was not possible at BV1. Also, as noted in Section 2.1.2.5 of this report, "operator depressurization before core damage" was not addressed in the back-end analysis because it already had been addressed in the front-end analysis. "Operator depressurization after core damage began" was outside the scope of existing emergency procedures and relegated to accident management considerations. Therefore, in-vessel recovery was not achieved at BV1. (Surry had an in-vessel recovery of 46.7% of the CDF.)

Through the CET top events, the BV1 IPE team was able to directly address phenomenological issues concerning hydrogen burn, direct containment heating, steam explosions, molten core concrete interactions, and steam/noncondensible gas pressurization. Although the members of the IPE team relied heavily on the results of the NUREG-1150 study of the Surry plant, they concluded that the late (and very late) hydrogen burn models for Surry were somewhat optimistic. For scenarios with sprays, the Surry analysts assumed that ignition would occur whenever hydrogen concentrations in the containment reached flammable levels. For scenarios without sprays, they assumed that ignition would occur as soon as recovery of sprays resulted in steam deinertion. In both cases, plentiful ignition sources of sufficient energy to ignite hydrogen were assumed to be present. The BV1 IPE team performed a MAAP analysis after eliminating the above nonconservative assumptions, but still found that late hydrogen burn had little effect on containment failure.

During the back-end containment walkthrough, the IPE team made a general inspection of the geometry and the "openness" of the containment and also inspected the location of potential ignition sources for combustible gases (page 4.1-2).

2.3.2 Dominant Contributors: Consistency with IPE Insights.

Table 2 of this report shows the results of SCIENTECH's comparison of the dominant contributors to the BV1 conditional failure probability with the results of the NUREG-1150 studies of the Zion and Surry plants and of the North Anna IPE. The CDF postulated for BV1 is the highest among the four plants listed in Table 2 and it is about five times higher than Surry's, the next highest in the list. However, as presented in Table 1-1 of the submittal, which is reproduced here as Table 3, the IPE team noted that the BV1 CDF was of the same order of magnitude as the CDFs derived from the PRAs of other PWRs based on comparable methods, databases, and work scopes. PLG performed all of the PRAs whose results are set out in Table 3, except for the one done of Surry, which was a NUREG-1150 study performed by NRC and its contractors.

Compared with the results obtained at the other three plants listed in Table 2, the BV1 results show a significantly lower percentage of containment intact, which is reflected in significantly higher percentages of early and late containment failures and containment isolation. A major reason for the higher containment failure percentage for BV1 is that the

Table 2. Containment Failure as a Percentage of Total CDF: Comparison with Other PRA Studies

Study	CDF (per rx year)	Early Failure	Late Failure	Bypass	Isolation Failure	Intact
North Anna IPE	6.8E-5	1.3	11.1	14.0	0.2	74.0
Zion/NUREG-1150	6.2E-5	0.5	24.0	0.5	1.0	73.0
Surry/NUREG-1150	4.1E-5	0.7	5.9	12.2	na	81.2
Beaver Valley 1 IPE	2.1E-4	6.5	43.4	4.5	16.3	29.3

na not available but included in early containment failure

Table 3. Comparison of PRA Results for Internal Events*

Plant	Mean CDF (per reactor-year)
Three Mile Island	4.4E-4
Midland	2.9E-4
Beaver Valley Unit 1	2.1E-4
Beaver Valley Unit 2	1.9E-4
Seabrook Station	1.7E-4
South Texas Project	1.7E-4
Diablo Canyon	1.3E-4
Surry (without cross-ties)	1.2E-4
Surry (with cross-ties)	0.4E-4

* Reproduced from the submittal [1]

BV1 back-end analysis does not take any credit for in-vessel recovery. Section 4.6.3.1, page 4.6-10 of the submittal, notes this as a major difference between the Surry and Beaver Valley Level 2 analyses.

In the results reported on the NUREG-1150 study of Surry, substantial credit was taken for the arrest of core damage before vessel breach (46.6% of the CDF for Surry involved no vessel breach). To arrest core damage, the ECCS injection must be restored and the core damage must not have progressed to a stage where the core has taken the form of an uncoolable geometry. For Surry, recovery of injection was due primarily to two events. First, for cases in which AC power was lost, it might be possible to recover injection once the power was restored. The BV1 IPE team argued that such a recovery of injection was unlikely mainly because, once core damage had begun, procedural guidance for operator actions to restore coolant flow would be unavailable. Second, in other types of accidents, the ECCS might be operating, but the RCS pressure would be so high that it would prevent injection into the vessel. As noted in the analysis of the Surry plant, depressurizing the RCS after initiation of core damage (uncovery of the top of the active fuel) could result:

- Because the pressurizer PORVs or SRVs were stuck open
- Through a temperature-induced RCP seal failure
- Because the operators deliberately opened the PORVs
- Through a temperature-induced SGTR
- Through a temperature-induced hot leg or surge line failure.

"Operator depressurization before core damage" was addressed in the plant model. The IPE team concluded that "operator depressurization after core damage began," (i.e., as described in the third bullet, above) should be considered in the analysis of accident management because it was outside the scope of existing emergency procedures. The conditions described in the remaining four bullets above were included in the BV1 CET, but only in the context of altering the RCS pressure at the time of vessel breach from what it was at the initiation of core damage.

As noted in Section 2.1.1, the BV1 IPE team defined early containment failure as that occurring before or within 4 hours of vessel breach. This definition is different from that used in the NUREG-1150 study, in which early containment failures of PWRs were defined as those occurring before or within a few minutes of vessel breach (page 2-13). Therefore, some containment failures that were considered early during the BV1 IPE would be binned as late under the NUREG-1150 definition. The BV1 IPE team used a 24-hour mission time in its analysis.

2.3.3 Characterization of Containment Performance.

The IPE team used a CET to characterize containment performance. The quantification of the CET included the following (Section 4.6.3, page 4.6-9):

- Determination of top event split fractions

- Combination of the split fraction values to determine conditional frequencies for each of the tree sequences

- Assignment of each CET sequence to a release category

- Summation of all CET sequences for each release category.

All of the above tasks, except for the determination of the CET top event split fractions, were performed using the event tree module of the RISKMAN computer program, which requires the following as input:

- Description of the top events in the CET

- Logic defining the structure of the tree

- Set of rules for assigning specific split fractions to account for dependencies on prior events

- Table of split fraction values

- Set of rules governing the assignment of CET sequences to each release category.

The above input parameters are described in detail in the submittal. The IPE team extracted most of the CET split fraction values directly from the results of the NUREG-1150 study of the Surry plant and, where necessary, used engineering judgment to quantify some of the fractions. The team members found that certain operator actions at Surry were beyond the scope of existing operating procedures at BV1, and, therefore, they conservatively ignored those actions in the back-end analysis, e.g., "operator depressurization of the primary system after the initiation of core melt." However, these actions were described well in the submittal, giving them visibility for later refinement or possible inclusion in accident management initiatives.

The BV1 IPE team applied the insights gained from MAAP calculations performed as part of the BV2 IPE in many accident sequences with both LOCA and transient initiators, including SGTR, and in various combinations of ECCS, auxiliary feedwater, and containment spray systems assumed to be failed. The specific insights used were ones regarding event timing, and included failures of the reactor coolant system (RCS) and steam generator tubes, ex-vessel hydrogen production and disposition, in-vessel fission product revaporization after vessel failure, and the transient inventory of water in the steam generator following SGTR.

Section 4.2.3 of the submittal describes three specific issues that were addressed regarding the BV1 containment: in-vessel and ex-vessel hydrogen generation, hydrogen combustion/detonation, and the effect of lead addition to the ex-vessel debris from the melting of a steel-clad lead shield. The team noted that significant uncertainties existed in the analysis of containment performance during severe accidents related to the generation of hydrogen and its disposition because BV1 has a relatively smaller subatmospheric containment than the other PWRs with atmospheric containments. For example, 40-percent

oxidation of the core zircalloy would raise the molar concentration of hydrogen in the dry air of the BV1 containment on the order of 10 percent, while 100-percent oxidation of zircalloy would raise the concentration to the order of 22 percent. A major uncertainty about in-vessel hydrogen generation was the result of core blockage caused by relocation of core materials. MAAP runs for fast SBO sequences at BV2 indicated that in-vessel hydrogen could be 75 percent higher if no blockage was assumed. The ex-vessel generation of hydrogen involved oxidation of the remaining metallic zircalloy, core structural steel, and steel rebar in the reactor cavity basemat. MAAP runs performed for BV2 showed substantial generation hydrogen (and carbon monoxide) as a result of steel rebar oxidation during core-concrete interactions.

The IPE team noted that the combustion/detonation potential of hydrogen produced during severe accidents was important in all phases of the accidents: before vessel breach, at vessel breach, shortly after vessel breach, and over the long term. The team found that several assumptions about hydrogen combustion/detonation made during the NUREG-1150 study of the Surry plant were not conservative. For example, in the analysis of Surry, a detonation limit of 14-percent hydrogen was assumed. However, experiments conducted on flame acceleration and deflagration-to-detonation of hydrogen-air mixtures at the Sandia National Laboratories FLAME facility showed this limit to actually be 12 percent. The BV1 IPE team used this lower limit. The IPE team assumed that hydrogen detonations would always fail the containment.

Containment failures at vessel breach due to HPME were due in part to hydrogen combustion. The pressure rise in vessel breach represented the combined effects of blowdown, hydrogen burning, DCH, and steam explosion. The IPE team assumed that large quantities of hydrogen would be available for combustion before and after vessel breach. If this hydrogen had not burned before or at vessel breach and the containment atmosphere was not inerted by steam, there was a high probability that a burn would occur in the "early" time frame, within 4 hours of vessel breach. Based on the lower flammability limit of 4% hydrogen in the air, the team assumed a probability of 0.965 that an "early" hydrogen burn would occur if containment sprays were operating and no high-pressure melt ejection (HPME) occurred at vessel breach. If debris was cooled and the containment sprays were operating, no significant source of additional hydrogen would be present in early time frame, and burns would to be unlikely. If an early hydrogen burn did occur, containment failure was assumed to be a certainty.

The IPE team assumed that if the debris was being not cooled, there was a 50-percent chance that there would be a hydrogen burn in the "late" time frame (more than 4 hours after vessel breach) if the containment was inerted, i.e., sprays were not operating. If the containment was not inerted (i.e., electrical power was available and sprays were operating) and debris was not cooled, it was assumed that late hydrogen burns would occur. If the debris was being cooled and the containment sprays were operating, no significant source of hydrogen would be present, and therefore late hydrogen burns would be unlikely.

Percentage contributions of hydrogen burn/deflagration-to-detonation transition to the various release classes were reported as follows (page 58, DLC response to the NRC RAI): [3]

- 0.2% of release category group (RCG) I - large, early containment failures and bypasses
- 0.0% of RCG II - small, early containment failures and bypasses
- 11.3% of RCG III - late containment failures.

2.3.4 Impact on Equipment Behavior.

Section 4.1.4 of the submittal describes the IPE team's assessment of the capability of BV1 equipment to survive in the harsh environments of severe accident conditions. This assessment was to ensure that equipment failures under severe accident conditions would not cause poor containment performance. Thus, the assessment was limited to assessing how a given accident could be mitigated once core damage had occurred; the potential for core damage arrest was analyzed in the Level 1 phase. Mitigation of a severe accident was deemed possible by cooling the damaged core and removing energy and radioactive materials from the containment atmosphere. Accident mitigation could be achieved at BV1 by operation of the containment spray and heat removal systems. The submittal describes the survivability of monitoring and actuation systems, containment spray systems, and the auxiliary feedwater system, as summarized below.

Monitoring and Actuation. In situations where an accident progresses to the point of severe core damage or beyond vessel breach, the actions still available to the operator to mitigate the consequences are reduced to those associated with understanding the condition of the containment and the performance and control of the remaining systems for core debris and containment cooling. The following are the conditions that provide important information to the operators after initiation of core damage and that rely on hardware located in the containment:

- RCS pressure
- Core exit temperature
- Coolant inventory
- Containment sump water level
- Containment pressure
- Containment area radiation
- Containment hydrogen concentration
- Containment atmosphere temperature
- Steam generator secondary-side water level

The IPE team relied on the IDCOR Technical Report 17, which addressed the issue of equipment survivability in a severe accident environment. In particular, the team used the results of the study of Zion, which is a PWR with a large, dry containment. These results are listed in Table 4.1-3 of the submittal.

Containment Spray Systems. Containment spray systems are the principal means of removing energy and radionuclides from the containment atmosphere. The BV1 containment has two spray systems: quench and recirculation. The quench spray pumps take suction from the residual water storage tank (RWST) and their operation is independent of containment conditions. For recirculation, the BV1 spray pumps were found not to be vulnerable to inadequate net positive suction head (NPSH) caused by containment temperature and pressure. Neither were these pumps found to be vulnerable to structural loads caused by potential containment displacements (at high pressures approaching containment failure).

Auxiliary Feedwater System. Under the conditions of a severe accident progression, survival of the auxiliary feedwater flow to the steam generators is important because it ensures heat transfer from the primary to the secondary side, which could significantly affect core degradation, containment integrity, and radionuclide behavior. Auxiliary feedwater pumps, which were assumed to operate during some Level 1 sequences, were found not to be vulnerable to containment expansion, nor to expected temperatures in the containment.

It appears that the issue of equipment survivability received adequate attention during the BV1 IPE in accordance with the level of detail requested in NUREG-1335.

2.3.5 Uncertainty and Sensitivity Analysis

The IPE team performed three sensitivity case runs. Sensitivity case A was to assess the relative importance of the in-vessel recovery after core degradation. This case was run using the dominant sequence model and by reducing the containment release to 50 percent. The net effect was a drop in the frequency of release category groups I and II by a factor of 2.

Sensitivity case B was run to determine the importance of thermally induced RCS failure before vessel breach. This case was run using guaranteed success split fractions, i.e., thermally induced RCS failures do not occur before vessel breach. The results showed a decrease in release category group II frequencies. The results also showed an increase in group I frequencies although induced steam generator tube rupture containment bypass frequencies were reduced. This increase was due to the higher probability of large, early containment failures resulting from the RCS remaining at higher pressures.

Sensitivity case C was a hand calculation to determine the impact of procedural modifications that would facilitate the deliberate depressurization of the RCS after core damage was initiated. This calculation was performed after changing the split fraction value for induced hot leg or surge line failure (IPS). The results showed that the implementation of procedures for deliberately depressurizing the RCS after core damage for fast station blackout TMLB'-type sequences would lower the group I frequency by 4.5 percent.

In addition, as described in section 2.3.1 of this report, the IPE team used the results of the MAAP sensitivity analysis performed for BV2, i.e., for the occurrence of potential hot leg or creep rupture.

2.4 Reducing The Probability of Core Damage or Fission Product Release

2.4.1 Definition of Vulnerability.

Table 6.3-1, page 6.3-6 of the submittal, lists BV1 vulnerabilities, which include four related to operator failures and three related to plant hardware failures. The IPE team identified these vulnerabilities by evaluating the major accident categories and the top-ranking sequences contributing to the BV1 CDF. Section 6.3.3 of the submittal describes two containment vulnerabilities identified by examining contributors to early containment release frequency: containment bypass and containment overpressurization.

The containment bypass resulted from interfacing system loss of coolant accidents (LOCAs) and steam generator tube ruptures (SGTRs). The contribution from interfacing system LOCAs to large, early releases of BV1 was 10.7% compared with 77% for the Surry plant. This contribution, which was lower than the one at Surry, resulted from there being only one interfacing system, the LHSI, located outside the containment at BV1. (The RHR system is located inside the containment.) The contribution from SGTRs to large, early releases of BV1 was 3.2% compared with 10% for the Surry plant. Thus, the combined contribution of interfacing system LOCAs and SGTRs to large, early releases of BV1 was 13.9% compared with 87% for the Surry plant. Bypass was considered a BV1 containment vulnerability. Changes to plant procedures and training were being implemented to enhance the operator response to such sequences. For "LOCA outside the containment," the IPE team identified the importance of improving guidance to the operators on the key valve to close.

Phenomena leading to containment overpressurization during a core-melt sequence included the RCS blowdown, early hydrogen burns, and DCH. The IPE team identified several actions that could lower containment overpressurization: lower the RCS pressure before vessel breach, flood the reactor cavity, and establish debris cooling after vessel breach. The team identified the importance of improving procedures to better perform the above mitigative actions. Using the diesel-driven fire system pump was recommended to inject water to the containment and to cool steam generator tubes during an SBO sequence. Throttling the quench spray pumps was recommended to conserve RWST inventory during an accident sequence.

It appears that the BV1 IPE team adequately identified back-end vulnerabilities and methods that could be used to improve containment performance under severe accident conditions.

2.4.2 Plant Improvements.

Back-end improvements that the IPE team stated would help reduce the BV1 containment bypass frequency and containment overpressurization failure frequency are described in Section 2.4.1 of this report. In addition, Section 6.4.1, page 6.4-1 of the submittal, describes an enhancement to update procedures to depressurize primary and secondary systems. The existing EOPs at BV1 were not explicit for sequences where high head safety injection (HHSI) was unavailable. Thus, the IPE team identified important plant enhancements to mitigate the progression of severe accidents at the BV1 plant.

2.5 Responses to Containment Performance Improvement Program Recommendations

One of the CPI Program recommendations that pertains to PWRs with large, dry containments is that utilities evaluate their containment and equipment vulnerabilities to hydrogen combustion (local and global) as part of their IPEs and that they identify the need for improvements in PWR procedures and equipment. Developing detonable mixtures of hydrogen on a global basis may be especially important for smaller subatmospheric containments like BV1. Consistent with these recommendations, the BV1 IPE team did the following:

- Visually inspected the containment geometry and openness and the location of potential ignition sources for combustible gases during the containment walkdown performed to gather information for the back-end analysis (Section 4.1-.2, page 4.1-2)

- Evaluated and found that the containment penetration seals were not vulnerable to thermal attack from hot combustion gases

- Addressed important issues regarding hydrogen generation and hydrogen combustion/detonation, as described in Section 2.3.3 of this report

- Analyzed the vulnerability of containment performance to hydrogen combustion using BV1 CET top events.

2.6 IPE Insights, Improvements, and Commitments

Figure 1, reproduced from the submittal, shows contributors to the BV1 CDF from sequences grouped by initiating event. The total CDF calculated for BV1 from internal initiators was 2.1[illegible]E-4 per reactor year. One of the important groups of sequences that drove the final results was loss of offsite power (23.9 percent of the total CDF) followed by loss of emergency AC power train (19.3 percent). The next three contributors in descending order of importance were the partial loss of the main feedwater (12.3 percent), total loss of river water
(11.2 percent), and non-isolable small LOCA (5.6 percent).

In order to gain more insights from the CDF results, the IPE team presented distributions that took into account particular conditions of the plant that depended not only on the initiating event of each accident sequence, but also on the response to one or more plant systems. Because each sequence could possess more than one of the given conditions, the resulting sequence groups were not always mutually exclusive. The IPE team presented the following results obtained for accident sequence classes of general interest (given as a percentile contribution to CDF): RCP seal LOCA, 46.6; SBO, 30.4; containment bypass/isolation failure, 20.7; loss of switchgear ventilation, 15.5; and ATWS, 20.1. A large fraction of the core damage was the result of a RCP seal LOCA, a significant fraction of which was caused by an SBO and by a loss of switchgear ventilation.

Figure 1

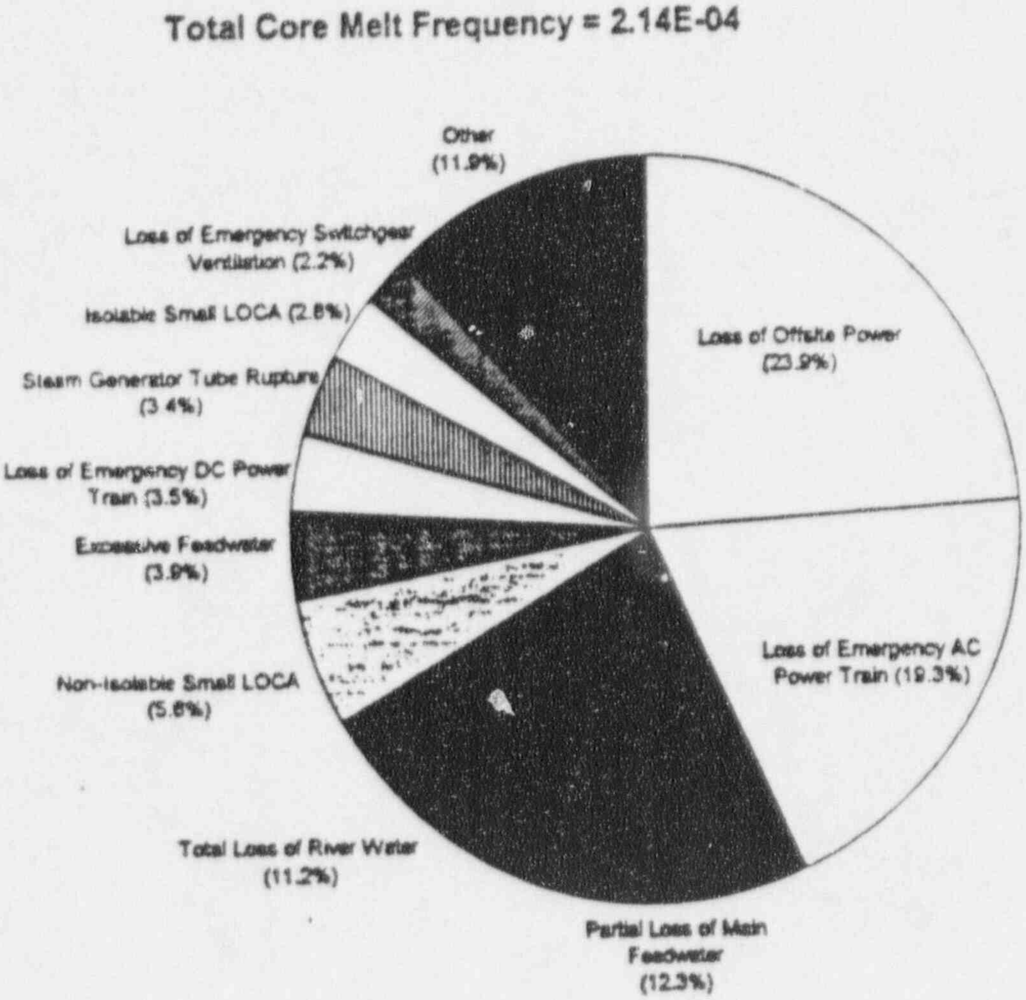

Figure 1. Contributors to CDF at BV1 from sequences grouped by initiating events.*

* Reproduced from the submittal [1]

Table 4, which is reproduced from Table 1-3, page 1.4-7 of the submittal, shows the characteristics of the primary system and of the containment at the initiation of core damage that are important for the accident progression. A major fraction of the CDF, i.e., 79 percent, was made up of sequences involving the primary system at high pressure (i.e., 17.6 percent at a pressure greater than or equal to 2,000 psia and 62 percent at a pressure between 600 psia and 2,000 psia). These results demonstrate the importance of depressurizing the primary system to be able to inject low-pressure coolant to cool the damaged core in-vessel or to reduce the possibility of HPME. But operator depressurization was not considered in the back-end analysis because operator actions after core damage were not proceduralized. Induced failure of the RCS hot legs and "PORV stuck open" were the only important depressurization modes.

Table 4. Plant Damage State Annual Frequency and Percentage of CDF*

RCS Pressure (psia)	Containment Bypassed Isolation/Not		Containment Not Isolated < 3 inch Leak	Small Bypass	Large Bypass	Total
	With Containment Heat Removal	Without Containment Heat Removal				
≥ 2000	3.01E-7 (0.1%)	2.99E-6 (1.4%)	3.40E-5 (15.9%)	4.01E-7 (0.2%)		3.77E-5 (17.6%)
600 - 2000	3.42E-5 (16.0%)	9.02E-5 (42.3%)	7.44E-7 (0.4%)	6.98E-6 (3.3%)		1.32E-4 (62.0%)
200 - 600	3.41E-6 (1.6%)	2.38E-7 (0.1%)	2.14E-8 (< 0.1%)	4.00E-9 (< 0.01%)		3.67E-6 (1.7%)
< 200	3.74E-5 (17.5%)	3.99E-7 (0.2%)	6.90E-8 (< 0.1%)	1.00E-6 (0.5%)	1.09E-6 (0.5%)	3.99E-5 (18.7%)
Total	7.53E-5 (35.2%)	9.38E-5 (44.0%)	3.48E-5 (16.3%)	8.42E-6 (4.0%)	1.09E-6 (0.5%)	2.13E-4 (100%)

* Reproduced from the submittal [1]

As shown in Table 4, at the initiation of core damage, "containment not isolated" (with openings of sizes less than 3 inches in diameter) accounted for 16.3 percent of the BV1 CDF and "containment heat removal systems not operating" accounted for 44.0 percent. The BV1 recirculation fan cooling system was not designed as a safety system and therefore was not used in the back-end analysis for containment heat removal. The status of the quench sprays system and of the recirculation sprays system were determined in the Level 1 model and they were not explicitly treated in the back-end analysis. No containment cooling (via the spray systems) recovery was modeled. Accident sequences that contributed 44 percent of the CDF involved no containment heat removal after the initiation of core damage.

3. CONTRACTOR OBSERVATIONS AND CONCLUSIONS

The BV1 IPE submittal contains a substantial amount of information regarding the requirements of Generic Letter 88-20, its supplements, and NUREG-1335, and appears to be complete in accordance with the level of detail requested in NUREG-1335. The IPE methodology used is described clearly in the submittal. The approach followed is consistent with the basic tenets of GL 88-20, Appendix 1, and the assumptions underlying the approach are clearly described. The important plant information and data are wel! documented and the key IPE results and findings are well presented. The submittal relies heavily on the results of the NUREG-1150 study performed for the Surry plant.

The BV1 IPE team used a definition for early containment failure that was more conservative than the one used in the NUREG-1150 study. For BV1, early containment failure occurred before or within 4 hours of vessel breach. In the NUREG-1150 study, it was defined as occurring before or within a few minutes of vessel breach.

The following is a BV1 containment feature that affected the results of the back-end analysis:

- The BV1 containment cavity stayed dry for most of the accident sequences. Even in those sequences where the cavity was flooded, the team argued that the containment sprays worked, making it less important that the aerosol was removed by the water pool overlying the melt. In characterizing the BV1 source terms, the team assumed that the cavity stayed dry during all of the accident sequences. (The presence of water in the cavity, however,was considered in calculating core-concrete interactions.)

The following are severe accident phenomena that affected the results of the BV1 back-end analysis:

- Early overpressurization of the containment accounted for 81.1 percent of the contribution to the early, large containment failure category (compared with 7 percent at the Surry plant). Direct containment heating caused by HPME strongly influenced early overpressurization failure of the containment. A major reason for the strong influence of HPME was a modeling assumption that did not allow for operator depressurization, which increased the likelihood of vessel breach at high pressure.

- Based on MAAP analysis performed during the BV2 IPE, the BV1 IPE team concluded that induced SGTR was not a concern for BV1, determining that hot leg or surge line failure had a much higher probability of occurring in sequences with RCS high pressure.

- BV1 has a relatively smaller subatmospheric containment than ther PWRs with atmospheric containments. This makes the issue of hydrogen combustion more important for BV1. For example, 40-percent oxidation of the core zircalloy would raise the molar concentration of hydrogen in the dry air of the BV1 containment on the order of 10 percent, while 100-percent oxidation of zircalloy would raise the concentration on the order of 22 percent.

The results of the back-end analyses at BV1 showed the following contributors to containment failure, given as a percentage of total CDF: early failure, 6.5; late failure 43.4; containment bypass, 4.5; containment isolation failure, 16.3; and intact containment, 29.3. A modeling assumption that had a major impact on these results was that there was no in-vessel recovery after the initiation of core damage. By comparison, the Surry plant had a 46.7 percent in-vessel recovery. Two of the containment characteristics that drove these contributors to containment failure were 1) the relatively small volume of the containment, making DCH and hydrogen burns more important, and 2) the high probability that the reactor cavity would be dry.

The 16.3-percent CDF that the team calculated for containment isolation failures is relatively high, e.g., 0.2 percent for the North Anna IPE and 1.0 percent for the Zion NUREG-1150 study. However, DLC has defined containment integrity conservatively: Containment isolation failure size was assumed to be less than 3 inches in diameter. Because the BV1 containment is subatmospheric, larger openings were not expected to exist. This definition appears to be more conservative than that used in many IPEs where the diameter of the opening for containment isolation was assumed to be greater than 2 inches. At BV1 isolation failures were not excluded based on small size and openings up to 1/8 inch in diameter were investigated. The isolation failures found at BV1 were 2 inches and 1 inch in diameter. During the IPES at other plants, an opening of diameter 2 inches or less did not cause containment isolation failure, whereas, at BV1, it did.

The IPE team identified two back-end plant vulnerabilities based on major contributors to large, early radionuclide releases. The first vulnerability involved phenomena leading to containment overpressurization during a core-melt sequence that included the RCS blowdown, early hydrogen burns, and DCH. The IPE team identified several actions that could lower containment overpressurization: lower the RCS pressure before vessel breach, flood the reactor cavity, and establish debris cooling after vessel breach.

The second vulnerability that the team identified was containment bypass from interfacing system loss of coolant accidents (LOCAs) and steam generator tube ruptures (SGTRs). Interfacing system LOCAs contributed 10.7 percent to the large, early releases of BV1 compared with 77 percent for the Surry plant. This relatively low contribution resulted from there being only one interfacing system, the LHSI, located outside the containment at BV1. (The RHR system is located inside the containment.) The SGTRs contributed 3.2 percent to the large, early releases at BV1 compared with 10 percent for the Surry plant. Thus, the combined contribution of interfacing system LOCAs and SGTRs to large, early releases at BV1 was 13.9 percent of large early, releases compared with 87 percent for the Surry plant. Changes to plant procedures and training were being implemented to enhance the operator response to such sequences. For "LOCA outside the containment," the IPE team identified the importance of improving guidance to the operators on the key valve to close.

Based on the review, SCIENTECH noted the following strengths in the BV1 IPE back-end analysis:

- In order to treat properly the dependencies between the front-end safety systems that are needed to prevent damage to the core, to the containment system, and to the support systems that tie both together, the IPE team included all the active containment systems (e.g., quench and recirculation sprays, containment isolation) in the front-end trees.

- Containment event tree development and quantification in the BV1 IPE is very thorough, well presented, and in accordance with the level of details requested in the GL 88-20 and NUREG-1335.

- It appears that the IPE team identified all relevant potential containment failure modes. All applicable containment failure modes that appear in Table 2-2 of NUREG-1335 were considered.

4. REFERENCES

1. Duquesne Light Company, "Beaver Valley Nuclear Station-Unit 1 Individual plant Examination (IPE) for Internal Events," October 1992.

2. American Nuclear Society, "Report of the Special Committee on Source Terms," September 1984.

3. Duquesne Light Company, "Forwards Request for Additional Information Re. GL 88-20," March 1995.

4. U.S. Nuclear Regulatory Commission, "Containment Performance Working Group Report," draft report for comment, NUREG-1037, May 1985.

APPENDIX
IPE EVALUATION AND DATA SUMMARY SHEET

PWR Back-End Facts

Plant Name

Beaver Valley 1

Containment Type

Large, dry, subatmospheric

Unique Containment Features

Fan coolers are not safety-related and therefore are not expected to work during severe accidents

Unique Vessel Features

None found

Number of Plant Damage States

143

Ultimate Containment Failure Pressure

126 psig (mean value)

Additional Radionuclide Transport and Retention Structures

Reactor building retention is credited.

Conditional Probability That The Containment Is Not Isolated

0.163

Important Insights, Including Unique Safety Features

The containment cavity consists of a lead shield that could melt and add to the core melt, thus changing core-concrete interactions; the cavity would stay dry for most of the accident sequences; Beaver Valley Unit 1 containment is similar to that of Surry Unit 1 and the Beaver Valley Unit 2

Implemented Plant Improvements

Changes to procedures and training that would reduce the frequencies of containment bypass and containment overpressurization failure and that would update procedures to depressurize primary and secondary systems are in progress or being reviewed.

C-Matrix

Information in the submittal was not sufficient to generate a C-matrix.

APPENDIX C

BEAVER VALLEY 1 NUCLEAR PLANT INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(HUMAN RELIABILITY ANALYSIS)