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TECHNICAL EVALUATION REPORT OF THE CALVERT CLIFFS INDIVIDUAL PLANT EXAMINATION BACK-END SUBMITTAL

FINAL REPORT

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Enclosure 3

E. EXECUTIVE SUMMARY

This Technical Evaluation Report (TER) documents the findings from a review of the back-end portion of the Baltimore Gas and Electric (BG&E) Individual Plant Examination (IPE) for the Calvert Cliffs Nuclear Power Plant (CCNPP).

E.1 Plant Characterization

CCNPP is a two-unit PWR of Combustion Engineering design, located on the western shore of Chesapeake Bay in Calvert County, Maryland. Each unit is housed in a large dry containment constructed of prestressed, post-tensioned reinforced-concrete, which is lined with steel on its inner face. In several aspects, such as RCS water volume, power, and core inventory, the design of CCNPP is somewhat similar to the Surry plant.

E.2 Licensee's IPE Process

The IPE back-end analysis was performed as a contractor-utility joint effort, with the project team consisting of the following: BG&E's Reliability Engineering Unit (REU), who provided input and support for the back-end analysis; Alfred Torri of Risk and Safety Engineering, who performed the majority of the back-end analysis and consulted on the front-end/back-end interface; BG&E's Nuclear Engineering Unit (NEU), who performed all MAAP calculations for the Level-2 analysis; EQE International, who performed the containment structural fragility analysis; Gabor, Kenton and Associates, who provided consulting services for the deterministic (MAAP) analyses; and Frank Hubbard of FRH, Inc., who also consulted on the back-end analysis. The IPE describes four independent reviews that were conducted for the CCNPP IPE. However, only one of these reviews, an in-house review conducted by BG&E's REU, pertained significantly to the back-end analyses.

The IPE back-end analysis starts with the binning of the results of the front-end analyses; i.e., a description of possible accident sequences, together with estimates of their annual frequencies of occurrence. A Plant-Damage State Matrix (PDSM) is constructed and used to assign core-damage sequences to Plant Damage State (PDS) bins, based on assignment rules that pertain to similarities in cut sets of core-damage scenarios. A Containment Phenomenological Event Tree (CPET) is constructed and quantified to model accident progression. Integrated into the CPET is the analysis of source-term characteristics, based on applicable MAAP analyses. A reduced set of (important) release categories, their frequencies, and a description of their associated radiological source terms are the end products of the CCNPP back-end IPE.

A limited number of sensitivity cases were conducted for the back-end evaluation of CCNPP. An uncertainty analysis was not performed as part of the study.

BG&E had a somewhat limited participation in the direction and execution of the back-end analysis, and the IPE clearly suggests that Alfred Torri (Risk and Safety Engineering) had the major role in overseeing and conducting technical analyses. A number of personnel in BG&E's

Reliability Engineering Unit likely have been assigned to acquire a detailed understanding of the back-end IPE, in order to maintain and use it in future plant decisions. The IPE provides only a very brief description of the Level-2 plant walkdown effort. The containment walkdowns (of both Units 1 and 2) were performed over a two-day period by a total of five team members (two BG&E personnel and three contractors). The walkdown involved a general visual inspection of the containment geometry and openness. The official freeze date of the IPE is March 19, 1992 (Section 2.4.3 of IPE).

E.3 Back-End Analysis

The submittal reports a Core Damage Frequency (CDF) of 2.4×10^{-4} per reactor-year due to internal initiators; 5.57% of this value is derived from internal flood events. This value of CDF are somewhat higher in comparison to CDF results obtained for other plants. Small LOCAs (21.44%), Loss of Offsite Power sequences (15.21%), internal flooding (5.57%), and a variety of plant transients (39%) are the leading contributors to core damage. Steam Generator Tube Rupture (1.88%) and Interfacing Systems LOCA (ISLOCA) sequences (0.77%) are the dominant bypass sequences.

PDSs were used to bin the end-states of the front-end analyses, and served as entry points to containment analyses. The PDS attributes include:

1. RCS pressure at core damage,
2. SI system status,
3. Containment isolation and bypass status, and
4. Containment heat removal and fission-product scrubbing (i.e., availability of sprays and fan coolers).

The PDS matrix resulted in the development of 366 possible combinations (PDSs) based on the foregoing attributes. After eliminating non-physical combinations, and bins having zero frequency, 56 PDSs were obtained. The set of 56 PDSs were further condensed to a smaller set of 15 Key Plant Damage States (KPDSs), in order to minimize the amount of CPET quantification that would be required.

Probabilistic quantification of severe accident progression for the key plant damage state bins was performed using the event tree methodology. The result of the CPET quantification process are accident progression sequences (i.e., CPET sequences) and their estimated frequencies of occurrence. A source-term analysis, including a source-term event tree (STET), is then used to develop release categories and their frequencies. The release categories are condensed to a set of key release categories, in a manner analogous to the development of key plant damage states from PDS bins.

The main CPET developed for Calvert Cliffs has fifteen top event nodes. CPET quantification, in a few instances (e.g., for ATWS sequences), relies on data developed for CET input quantification in the NUREG-1150 analysis of Surry. In other cases, CPET split-fraction quantification is performed using results from plant-specific MAAP analyses. Most of the important phenomena of interest to PWR severe accident phenomenology, including hot leg or steam generator tube rupture due to natural circulation; in-vessel core coolability; hydrogen generation and combustion; in-vessel steam explosions; high pressure melt ejection; direct containment heating; molten core-concrete interactions; containment failure in the early and late phases of the accident; hydrogen combustion in the late phase of the accident; core debris coolability; and basemat melt-through are all treated in the CPET.

The following aspects of the CCNPP IPE back-end analyses were based directly on the Surry analyses: (1) probability of containment failure during ATWS sequences; (2) CET quantification for ATWS scenarios; (3) RCS failure modes at high pressure; (4) vessel melt-through time delays; and (5) source term development for large early containment failure.

The CPET analysis for CCNPP resulted in 444 CPET sequences for each of the 15 KPDSs that served as initial conditions to the analysis. These 444 CPET sequences are binned into source term release categories based on a Source Term Event Tree (STET). The STET analysis is analogous to the PDS binning of Level-1 sequences. The result of the STET is a set of 36 release categories. The source term binning is based upon the following attributes:

1. Debris quenched and cooled in-vessel,
2. Containment failure time,
3. Containment failure mode,
4. Configuration of debris in cavity, and
5. RCS pressure at early vessel melt-through.

Results from the containment analyses indicate that, upon core damage, the conditional probability of radiological releases (including containment bypass events) for CCNP is 0.52. The biggest contributors, by initiating event to the early releases are station blackout sequences and small break LOCA sequences. The largest contributor to late releases are the station blackout sequences.

Table E.1 summarizes the results of the CCNP IPE analyses leading to containment failure.

Table E.1 Containment Failure as a Percentage of Total CDF

Containment Failure Mode	Conditional Probability (Frequency Per Reactor Year)
Early Failure	0.086 (1.9×10^{-5})
Late Failure	0.4 (8.9×10^{-5})
Bypass (V)	10^{-4} (3.5×10^{-8})
Bypass (SGTR)	0.031 (7.0×10^{-6})
Intact	0.486 (1.1×10^{-4})

E.4 Containment Performance Improvement (CPI) Issues

Generic Letter 88-20, Supplement Numbers 1 and 3 [7-8] identified specific Containment Performance Improvements (CPIs) to reduce the vulnerability of containments to severe accident challenges. For PWRs with large dry containments, it is requested that the licensee evaluate the IPE results for containment and equipment vulnerabilities to hydrogen combustion (local and global), and determine the need for procedural and/or hardware improvements. The licensee has analyzed the vulnerability of the CCNP containment to the potential for hydrogen pocketing [7]. The licensee noted that all compartments in the lower containment which contain the primary system components, are open at the top, communicating with the upper containment free volume. Thus, hydrogen released from the primary system can rise to the upper compartment. The only exception is the pressurizer cubicle, where there is a region about 5.5 feet in height (cross-section of 12.5 feet x 15 feet) above the pressurizer, which is bound by concrete walls and a ceiling. This region communicates to the upper compartment of the containment through an opening 2 feet x 5 feet in cross-section. Inside the cubicle, above the pressurizer, there are PORVs, associated block valves, piping, instrumentation, and electrical junction boxes. It is possible for hydrogen to accumulate in the top 5.5 feet of the cubicle (approximately 1000 ft^3 in volume), only if a LOCA occurs in the piping inside the cubicle. However, since the length of the RCS piping inside the cubicle is small, and the possibility of a LOCA in the piping is rather remote. Ignition sources exist in the cubicle, and the licensee believes that the ignition sources will ignite the hydrogen before an explosive accumulation occurs. Under the remote possibility of a Deflagration-to-Detonation Transition (DDT) in the cubicle, a shock wave will propagate to the metal wall of the cubicle. The sheet metal wall is designed to fail at 5 psid. The failure of the wall is stated to prevent the reminder of structure from failure. Since the sheet metal wall faces the inner portion of the upper containment, the licensee believes that blowout of the wall will not lead to projectiles that can fail the containment.

Global hydrogen combustion was considered in the IPE submittal, and the conditional probability of containment failure at vessel breach due to hydrogen combustion was assessed to be in the range of 0.0015 to 0.033 (for different KPDSs). However, the conditional probability of containment failure due to detonation was assessed to be of the order of 10^{-4}.

Equipment vulnerabilities to conditions caused by hydrogen combustion was considered in the CPET. The only equipment which impact the results, and are taken credit for in the back-end analyses, are the air coolers and the containment sprays. The CET top event CA considers the availability of the fan coolers and the sprays for long term heat removal. For the containment spray system, the containment spray headers, several manual valves, a check valve and an air-operated valve are located inside the containment. The licensee concludes that these components are not vulnerable to hydrogen combustion, and the conditional probability of failure of the spray system, was assigned a value of 0.02. Failure of air coolers is more probable because air cooler power cables, control cables, fans, and motors are located inside the containment. A split fraction of 0.1 was assigned to the failure of air coolers due to combustion.

Three accident sequences involve failure of containment cooling systems after a hydrogen combustion, and thus leading to containment failure. These sequences correspond to a conditional probability of containment failure of approximately 2.2% of the CDF.

E.5 Vulnerabilities and Plant Improvements

The licensee does not provide an explicit definition used to identify a "vulnerability" as pertaining to containment analyses. Plant improvements, however, were identified and implemented in response to a plant-specific vulnerability assessment. These plant improvements related to insights that were derived from the front-end analyses. Insights derived from the back-end analyses did not result in implementation of plant improvements, but did produce some observations concerning accident management considerations and evaluation of phenomenological uncertainties. These are summarized in the IPE (Section 4.8), and reproduced in Section 2.4.2 of the TER. The IPE states that these insights will be considered in the development of CCNPP Severe Accident Management guidelines. It appears that no plant improvements will result from the containment analyses.

E.6 Observations

The important points of the technical evaluation of the Calvert Cliffs IPE back-end analysis are summarized below:

- The Back-End portion of the IPE supplies a substantial amount of information with regards to the subject areas identified in Generic Letter 88-20, and NUREG-1335. For the most part, the separate models used in the Calvert Cliffs IPE Back-End analysis are technically sound. Most sections of the IPE are well written and concise.

- The licensee has addressed all phenomena of importance to severe accident phenomenology in PWRs. The licensee has also recognized the importance containment failure modes relevant to CE plants, such as vessel rocketing and cavity failure at vessel breach.

- The treatment of phenomenological issues in the CPET appears reasonable, and a good balance is struck between use of plant-specific results, generic results, and use of engineering judgment.

- The submittal has addressed the recommendations of the CPI program (GL 88-20, Supplements 1 and 2).

- The conditional probability of early containment failure at CCNP (due to overpressurization, vessel rocketing, etc.) is 3.6%, and is approximately a factor of five larger than that calculated in the NUREG-1150 analyses for Zion or Surry. The differences in results can be attributed to a rather conservative treatment of DCH in the CCNP IPE submittal. Additionally, the lack of penetrations in the lower vessel head and the configuration of cavity at CCNP leads to the possibility of containment failure due to rocketing, and due to cavity failure at vessel breach. Identification of these failure modes have contributed to the calculation of a larger value for the conditional probability of early containment failure.

- The conditional probability of late containment failure at CCNP (due to overpressurization and combustion of non-condensible gases generated by MCCI) is 40%, and is larger than that calculated for Zion (24%) or Surry (5.9%). It is assumed in the submittal that the unavailability or failure of containment heat removal after core damage, always leads to late containment failure, if containment has not failed earlier. No credit is given for the recovery of AC power.

TABLE OF CONTENTS

LIST OF TABLES

NOMENCLATURE

AC	Alternating Current
AFW	Auxiliary Feedwater
AOP	Abnormal Operating Procedure
AOV	Air Operated Valve
ATWS	Anticipated Transient Without Scram
BG&E	Baltimore Gas and Electric Company
CAC	Containment Air Cooler
CCI	Core Concrete Interactions
CCNPP	Calvert Cliffs Nuclear Power Plant
CCPRA	Calvert Cliffs Probabilistic Risk Assessment
CCW	Component Cooling Water
CDF	Core Damage Frequency
CRD	Control Rod Drive
CET	Containment Event Tree
CHR	Containment Heat Rejection
CIS	Containment Isolation Signal
CPET	Containment Phenomenological Event Tree
CPI	Containment Performance Improvement
CS	Containment Spray
CSS	Containment Spray System
CVET	Cavity Event Tree
DC	Direct Current
DCH	Direct Containment Heating
DET	Decomposition Event Tree
DF	Decontamination Factor
DG	Diesel Generator
ECCS	Emergency Core Cooling Systems
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
EPS	Electric Power System
ESF	Engineered Safety Features
ESFAS	Engineered Safety Features Actuation System
FMEA	Failure Modes and Effects Analysis
FPS	Fire Protection System
FPW	Fire Protection Water
GI	Generic Safety Issue
GL	Generic Letter
HEP	Human Error Probability
HPME	High Pressure Melt Ejection
HPSI	High Pressure Safety Injection
HRA	Human Reliability Analysis
HVAC	Heating, Ventilation, and Air Conditioning

NOMENCLATURE (Continued)

IAS	Instrument Air System
IN	Information Notice
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
ISLOCA	Interfacing Systems Loss of Coolant Accident
KPDS	Key Plant Damage State
KRC	Key Release Category
LOCA	Loss of Coolant Accident
LOSP	Loss of Offsite Power
LT-SBO	Long Term Station Blackout
MAAP	Modular Accident Analysis Program
MCC	Motor Control Center
MFW	Main Feedwater
MOV	Motor Operated Valve
NEU	Nuclear Engineering Unit, Baltimore Gas and Electric Company
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NSW	Normal Service Water
PCS	Power Conversion System
PDS	Plant Damage State
PDSM	Plant Damage State Matrix
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
RC	Release Category
RCS	Reactor Coolant System
REU	Reliability Engineering Unit, Baltimore Gas and Electric Company
RFP	Reactor Feedwater Pump
RHR	Residual Heat Rejection
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RSS	Reactor Safety Study
RWT	Refueling Water Tank
SBO	Station Black-Out
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SOP	System Operating Procedure
SORV	Stuck-Open Relief Valve
SRV	Safety Relief Valve
SSW	Station Service Water
STET	Source-Term Event Tree
TER	Technical Evaluation Report

NOMENCLATURE (Continued)

TW	Loss of Decay Heat Removal
USI	Unresolved Safety Issue
VAC	Volts Alternating Current
VDC	Volts Direct Current
VMT	Vessel Melt-through

1. INTRODUCTION

This Technical Evaluation Report (TER) documents the results of the review of the Calvert Cliffs Nuclear Plant (CCNP) Individual Plant Examination (IPE) Back-End submittal [1]. This TER 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 determine if the IPE submittal essentially provides the level of detail requested in the Submittal Guidance Document, NUREG-1335,
- To assess the strengths and the weaknesses of the IPE submittal,
- To provide a preliminary list of questions based on this limited review, and
- To complete the IPE Evaluation Data Summary Sheet.

The remainder of Section 1 of this report describes the technical evaluation process employed in this review, and presents a summary of the important characteristics of the Calvert Cliffs nuclear plant related to containment behavior and post-core-damage severe accident progression, as derived from the IPE. Section 2 summarizes the review technical findings, and briefly describes the submittal scope as it pertains to the work requirements. Each portion of Section 2 corresponds to a specific work requirement as outlined in the NRC contractor task order. A summary of the overall IPE evaluation and review conclusions are summarized in Section 3. Section 4 contains a list of cited references. Appendix A to this report contains the IPE evaluation data summary sheets.

1.1 Technical Review Process

The technical review process for back-end analysis consists of a complete examination of Sections 1, 2 and Sections 4 to 6 of the IPE. In this examination, key findings are noted; inputs, methods, and results are reviewed; and any issues or concerns pertaining to the submittal are identified. The primary intent in the review is to ascertain whether or not, or to what extent, the back-end IPE submittal satisfies the major intent of Generic Letter (GL) 88-20 [3] and achieves the four IPE sub-objectives. In addition, the submittal is evaluated with respect to the four review objectives listed above, including completeness with respect to IPE guidelines. A list of questions and requests for additional information was developed to help resolve issues and concerns noted in the examination process, and was forwarded to the licensee. The licensee responses [9] were reviewed. The final TER is based on the information contained in the IPE submittal [1] and the licensee responses to the NRC Requests for Additional Information (RAIs) [9].

1.2 Containment Analysis

CCNPP is a two-unit PWR with a large, dry containment. Each unit consists of a two-loop, 2700 MW(t) Pressurized Water Reactor (PWR) of Combustion Engineering design. The containment consists of a post-tensioned, pre-stressed reinforced concrete cylindrical wall and shallow dome roof. The entire containment structure is lined with 1/4 inch thick ASTM A36 steel that provides an essentially leak-tight barrier for the containment. The containment structure is monolithically connected to, and founded on, a 10 feet thick reinforced-concrete foundation slab. The plant is located on the western shore of Chesapeake Bay in Calvert County, Maryland.

The important containment design characteristics and features at CCNPP include the following items:

- Reactor Vessel and Cavity Configuration A unique feature of CCNPP is that the bottom of the reactor vessel is approximately level with the containment floor elevation. There are no penetrations through the bottom of the vessel. In addition, there is a 30 square inch opening through the biological shield that provides access to the bottom of the reactor vessel. This opening allows water to flow freely into the cavity, and also provides a path for molten debris to spread over the containment floor.

  An important impact of this configuration on severe accident progression is that vessel melt-through may be considerably delayed for core damage scenarios that lead to a significant water pool on the containment floor. This is because the water elevation will be above the bottom of the vessel head, thus acting to cool the bottom portion of the molten debris in-vessel. The potential for a severe ring/circumferential failure of the vessel (above the submerged level) thus becomes a concern.

- Containment Floor Configuration. With the exception of a slight depression under the reactor vessel, the containment floor is essentially flat (sloped slightly toward drains). This floor configuration, together with the configuration of the cavity access opening, helps ensure smooth flow and even spreading of core debris for low pressure vessel breach scenarios.

- Containment Sumps. The containment sumps consist at CCNPP consist of a normal sump and an emergency sump. The normal sump is 3' 6" deep, is not protected by a curb, and it drains to one (of two) ECCS pump room in the Auxiliary Building basement. The emergency sump is located next to the normal sump, is protected by a 16" high concrete curb (topped with a steel grating), and drains to both ECCS pump rooms in the Auxiliary Building basement.

Most of these above features are modeled specifically in the CCNPP back-end analysis, and some have an important impact on the Level-2 analysis.

Tables 1 and 2 provide a comparison of containment features between the Calvert Cliffs plant, Surry plant (a Westinghouse PWR with a sub-atmospheric containment), and Zion (a Westinghouse PWR with a large, dry containment). It can be seen that the containment features are comparable between CCNP and the Surry plant. Table 1 shows that the ratio of containment free volume-to-power is somewhat larger for Calvert Cliffs than for Surry; the mass of zircaloy-to-free volume ratio and the maximum hydrogen concentration are also higher than in Surry and Zion, and is typical of Combustion Engineering plants.

Table 1 Summary of Key Plant and Containment Design Features for the Calvert Cliffs Plant

Feature	Calvert Cliffs	Surry	Zion
Power Level, MW(t)	2,700	2,441	3,236
Volume of RCS Water, m^3	313	283	368
Free Volume of Containment, m^3	56,633	46,440	81,000
Mass of Fuel, kg	97,732	79,652	98,250
Mass of Zircaloy, kg	25,852	16,466	20,230
RCS Water Volume/Power, m^3/MW(t)	0.12	0.12	0.11
Containment Volume/Power, m^3/MW(t)	21	19	25
Zr Mass/Containment Volume, kg/m^3	0.45	0.35	0.25
Fuel Mass/Containment Volume, kg/m^3	1.7	1.7	1.2
Maximum Quantity of H_2 Generated by Zirconium Oxidation, kg	1134	720	890
Maximum H_2 Concentration, 10^{-3} moles/m^3	11.6	9	6

Table 2 Comparison of Containment Capacities

Feature	Calvert Cliffs	Surry	Zion
Design Pressure	47 psig (4.2 bars)	45 psig (4 bars)	47 psig (4.2 bars)
Failure Pressure	117 psig (9 bars)	126 psig (9.8 bars)	134 psig (10.2 bars)
Concrete Type	Limestone	Basaltic	Limestone

2. CONTRACTOR REVIEW FINDINGS

The present audit has assessed the Calvert Cliffs IPE submittal with respect to the recommendations of the Generic Letter (GL) 88-20, and the guidance provided in NUREG-1335. The responses of the licensee were also reviewed. The findings of the present review are reported in this section, and follow the structure of Task Order Subtask 1.

2.1 Licensee's IPE Process

2.1.1 Completeness and Methodology

The IPE submittal contains a substantial amount of information in accordance with the recommendations of GL 88-20 and NUREG-1335. The methodology employed in the Calvert Cliffs IPE submittal for the back-end evaluation is clearly described, and the IPE is consistent with GL 88-20. The back-end evaluation starts with the results of the front-end analyses; i.e., a description of possible accident sequences leading to core damage, together with estimates of their annual frequencies of occurrence. A plant-damage state matrix (PDSM) is constructed and used to bin core-damage sequences having similar cut sets into given plant damage state (PDS) bins. A set of assignment rules (Table 4.3.6 of the submittal) based on the cut-set attributes of core-damage sequences is used as the basis for constructing the PDSM. 166 unique plant damage states are modeled in the IPE; 56 of these were quantified with non-zero frequency. These PDSs are condensed to a set of 15 key plant damage states (KPDS) used subsequently in the back-end analysis. The procedure for condensing to KPDSs does not rely exclusively on frequency truncation, but rather, also on conservative application of a PDS severity ranking scheme. A containment phenomenological event tree (CPET) is then constructed and quantified to model severe-accident progression. The CPET is applicable to all KPDSs, even though split fractions used for quantification of the CPET are different for various KPDSs. 15 top events are used to describe the CPET (in comparison to 71 for the NUREG-1150 CET analysis of plant Surry). Analyses that support the construction of the CPET (and the development of split fractions) are described; however, logic of the CPET top nodes is not generally presented in the back-end IPE. A source-term event tree (STET) is constructed to assign each CPET sequence to a given release category. A total of 36 release categories result from this assignment. For each KPDS, a MAAP analysis is performed to evaluate split fractions for use in CPET quantification. A best-estimate of a single representative sequence is constructed as the basis for each MAAP analysis. CPET quantification for all KPDSs results in an assessment of release category (RC) frequencies. The 36 RCs are condensed to a set of 6 key release categories (KRC) based on a binning procedure. In general, the radiological source term associated with each RC is based on the results of MAAP analyses for representative severe-accident sequence scenarios. The results of the CCNPP back-end analysis are descriptions of RCs and KRCs and evaluations of their frequencies of occurrence, and their associated source terms.

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

CCNPP is a two-unit nuclear power plant with a Combustion Engineering Nuclear Steam Supply System (NSSS). Only Unit-1 was modeled in the IPE. The IPE submittal notes that an effort was made to identify any major differences between Units 1 and 2 that would effect the applicability of the IPE results. Although some differences related to the Level-1 IPE were mentioned in the submittal, apparently, no major differences were found that were judged to alter the applicability of the Unit-1 back-end results to Unit 2. (A limited walkdown of the Unit-2 containment was undertaken to verify similarities in configuration and geometry with the Unit-1 containment).

The official freeze date of the IPE is March 19, 1992 (see Section 2.4.3 of IPE). The containment walkdowns were conducted on June 20-21, 1991. Plant-specific information for the IPE was obtained from a number of controlled plant documents. The IPE submittal notes that plant walkdowns were conducted to support the use of the controlled references to ensure that the CCNPP IPE represents the as-built, as-operated plant condition.

2.1.3 Licensee Participation and Peer Review of IPE

The IPE back-end analysis was performed as a contractor-utility joint effort, with the project team consisting of the following: BG&E's Reliability Engineering Unit (REU), who provided input and support for the back-end analysis; Alfred Torri of Risk and Safety Engineering, who performed the majority of the back-end analysis and consulted on the front-end/back-end interface; BG&E's Nuclear Engineering Unit (NEU), who performed all MAAP calculations for the Level-2 analysis; EQE International, who performed the containment structural fragility analysis; Gabor, Kenton and Associates, who provided consulting services for the deterministic (MAAP) analyses; and Frank Hubbard of FRH, Inc., who also consulted on the back-end analysis. The IPE describes four independent reviews that were conducted for the CCNPP IPE. However, only one of these reviews, an in-house review conducted by REU, pertained significantly to the back-end analyses. No significant review comments related to the back-end analyses, appears to have been obtained.

2.2 Containment Analysis

This section provides a review of PDS binning, CET analyses, release category definitions, severe accident analyses, and the containment structural analyses in the submittal.

2.2.1 Front-End/Back-End Dependencies

The results of the front-end event trees in the CCNPP IPE are accident sequences and their frequencies. The CCNPP IPE submittal reports a CDF, after enhancements, equal to 2.4×10^{-4} per reactor-year due to internal initiators; 5.57% of this value is due to internal flooding. Before enhancements, the estimated CDF was 3.2×10^{-4} per reactor year. Both these values of CDF are high in comparison to CDF results obtained for most other plants. The IPE documentation is said to be based on information and insights derived from the analysis producing the 3.2×10^{-4} per year CDF result; however, most quantitative results of the back-end analysis suggest a core-damage frequency of 2.23×10^{-4} per reactor year. Of the total internal-events CDF, 21.44% is due to small LOCAs; 15.21% is due to loss of offsite power; 5.97% is due to plant trip; 5.57% is due to internal flooding; 5.12% is due to loss of 120V DC system; 5.09% is due to loss of control-room ventilation; 5.06% is due to loss of 480V AC system; 4.54% is due to Loss of Component Cooling Water (CCW) system; 4.27% is due to loss of 120V AC system; 3.65% is due to loss of one 13 KV or 500 KV bus; 3.24% is due to Loss of Main Feed Water (MFW); 2.53% is due to spurious PORV opening; 1.88% is due to Steam Generator Tube Rupture (SGTR); 0.77% is due to Interfacing Systems LOCA (ISLOCA); and 15.66% is due to other initiating events.

The entry points to the containment event trees are the plant damage states. PDSs are groupings of Level 1 core damage sequences, based on similarities in accident progression and availability of containment safeguards. In the CCNP IPE submittal, core-damage cutsets from the Level-1 analysis are grouped into similar plant damage states (PDSs), using a plant damage state matrix derived from PDS assignment rules. The following steps were employed in developing plant damage states: (1) selection and coding of plant damage state parameters; (2) development of a plant damage state matrix based on assignment rules that depend on sequence cutset characteristics; (3) application of the PDS matrix to develop PDS bin frequencies; and (4) condensation of the plant damage states to a set of key plant damage states based on PDS severity ranking.

Selection and Coding of Plant Damage State Parameters. Before PDSs were defined, plant systems, conditions, and features that could have a significant effect on the potential course of an accident were identified. These fell under one of the following three categories: RCS pressure at vessel melt-through, availability of water in the reactor cavity at melt-through; and containment safety features availability (including isolation/bypass status, containment sprays status, fan coolers status, potential for fission-product scrubbing). Each cutset from Level 1 was examined, and information important to the CPET analysis was identified and coded. The sequence, initiating event, and cutset (basic event) were used as inputs to this process. Section 4.3.5.1 of the IPE presents the PDS designator coding scheme used in the IPE to describe the cutset and sequence information required for subsequent analysis. Four attributes (each designated by a single character field) were developed for this coding: (1) RCS pressure at core damage (3 categories); (2) safety injection (SI) system status (8 categories); (3) containment isolation and bypass status (6 categories); and (4) containment heat removal and fission-product scrubbing mechanisms (3 categories).

The Level-1/Level-2 interface analysis of the CCNPP IPE does not state that a truncation limit was applied to core-damage sequences passed through to the Level-2 analysis. Hence, it is inferred that all Level-1 sequences are initially involved in the PDS coding and binning process. As discussed below, truncation limits are applied to the PDS-bin frequencies, which are interpreted in the IPE as functional sequence frequencies.

Development of PDS Matrix. The inputs to this second phase of the Level-1/Level-2 interface analysis were the PDS attribute codes and attribute definitions described above. The result of this phase was a matrix that categorizes PDS codes in terms of each combination of possible PDS attributes. The possible 4-character code combinations derived from the PDS matrix represents the complete set of possible plant damage state bins. The PDS attributes used to develop the PDS bins are listed below:

1. RCS pressure at core damage (a---). The following three ranges of RCS pressure at core damage are considered in the CCNPP interface analysis:

   - Pressure < 200 psia (a=L); low pressure
   - Pressure between 200 and 1,300 psia (a=M); moderate pressure
   - Pressure > 1,300 psia (a=H); high pressure

2. SI system status (-b--). This PDS attribute is used to define the extent and timing of cavity flooding, based on quantity and timing of water injection from the Refueling Water Tank (RWT). This element of plant status is important in determining the potential for, and nature of, debris cooling. The following eight values of SI system status are considered in the interface analysis:

   - Containment Spray (CS) and High Pressure Safety Injection (HPSI) pumps inject before Vessel Melt-Through [VMT] (b=G)
   - CS Pumps inject before VMT (b=B)
   - HPSI pumps inject before VMT (b=C)
   - HPSI pumps inject after VMT (b=L)
   - CS and HPSI pumps inject after VMT (b=H)
   - CS pumps inject after VMT (b=A)
   - Only the RCS volume is on the containment floor (b=R)
   - No water on the containment floor (no RWT injection) (b=N)

3. Containment isolation and bypass status (--c-). This PDS attribute is used to determine the nature of radioactive release and containment failure. The following five situations are considered:

- Containment isolated and not bypassed (c=I)

- Containment not isolated or failed prior to core damage; leak area smaller than or equivalent to a 4" diameter hole (c=S)

- Containment not isolated or failed prior to core damage; leak larger than the area of a 4"-diameter hole (c=W)

- Containment not isolated due to failure to isolate the containment sump drain line (c=D)

- Small containment bypass (less than or equal to the area of a 4"-diameter hole); i.e., SGTR or small V-sequence (c=B)

- Large containment bypass (greater that the area of a 4"-diameter hole); i.e., large V-sequence (c=V)

An event-tree is constructed in the CCNPP IPE for determining the leak size for V sequences (see Figure 4.3.7.1 of IPE).

4. Containment heat removal and fission-product scrubbing (---d). The mechanisms for containment heat removal and fission-product scrubbing are interrelated. These items are important for the potential containment failure mode and radiological source term. The following three cases are considered in the interface analysis for this PDS attribute:

   - Containment heat removal and fission-product scrubbing is provided by the operation of the containment spray system and the recirculation system (d=P)

   - Containment heat removal occurs without significant fission-product scrubbing, by the operation of the containment fan air cooler system. If the status of containment isolation and bypass is c=W or c=V, the status of this PDS is non-consequential and is ignored. (d=O)

   - Neither the containment spray nor the air cooler system is available (d=F)

The PDS matrix resulted in the development of 366 possible combinations (PDSs) based on the above-mentioned attributes. Some of these initial PDSs represented combinations of PDS parameters which were not possible, due to physical constraints or due to the structure of the plant model. The PDS analysis produces a total of 166 possible PDSs. (Note that Section 4.3.5.2 of the IPE states that 166 PDSs result, however, a count of PDSs based on the PDS matrix summarized in Table 4.3.5 indicates that 164 PDSs are obtained).

PDS assignment rules used to bin each Level-1 sequence into an appropriate PDS are presented in Table 4.3.6 of the IPE. These rules were used to describe how the successes or failures of different top events and initiating events were combined, using Boolean logic, to produce an outcome (letter assignment) for each of the PDS attributes.

Application of PDS Matrix. The PDS assignment rules were applied to each Level-1 sequence in order to define the PDS attributes of each sequence. Sequences that possessed attributes common to a particular PDS bin (i.e., with combinations of PDS attributes that placed it into a particular element of the PDS matrix) were assigned to that PDS bin, and the frequency of that PDS bin was appropriately accumulated. The result is an assessment of total frequency for each PDS bin; the sum of PDS-bin frequencies being equal to the total internal-events CDF. From this process, 56 PDS bins having non-zero frequency were obtained.

Condensation to Key Plant Damage States. The set of 56 PDSs were condensed to a smaller set of 15 Key Plant Damage States (KPDSs). This was accomplished to minimize the amount of CPET quantification that would be required. The CPET should be quantified for each PDS based on appropriate values for split fractions that are obtained from MAAP parameter runs that describe the PDS. PDS condensation was introduced to reduce the number of MAAP analyses required to develop split fractions. For the smaller set of KPDSs, an accident sequence was constructed which represents the PDSs in that KPDS bin, and a MAAP analysis was conducted based on this representative sequence.

Section 4.3.7 of the IPE describes the details of the procedures used to condense PDSs to KPDSs, and to obtain resulting KPDS frequencies. Truncation limits are first applied to the PDS frequencies. A truncation limit of 10^{-7} was applied to all sequences except bypass sequences where a truncation limit of 10^{-8} was applied. Thus, the sum of KPDS frequencies does not equal the total CDF. In addition to frequency truncation, the condensation procedure itself lumps PDSs into KPDS bins in a conservative manner. The KPDSs are selected and derived such that a lower-frequency PDS with a lower consequence is always binned into a higher-frequency PDS with a higher consequence. Nine steps were followed in the condensation process. 15 KPDSs were developed to ensure that the PDS used to represent a KPDS contributed no less than 50% to the total frequency of that KPDS.

A brief description of the 15 KPDSs developed as a result of the CCNPP IPE Level-1/Level-2 interface analysis is provided below:

1. HHIP: Total loss of main feedwater; one train of HPSI and CS are available and continue after recirculation switchover; containment is isolated and cooled by two CACs.

2. MBIO: Small break LOCA inside containment; HPSI is not available; two trains of CS are available, but fail at recirculation switchover; the containment is isolated and cooled by two CACs.

3. HRIF: Station blackout with total loss of main feedwater; neither HPSI or CS is available; containment is isolated, but no CACs are available.

4. MCIF: Small LOCA inside containment; two trains of HPSI are available, but fail at recirculation switchover; CS not available; containment is isolated, but no CACs are available.

5. MRIO: Small LOCA inside containment; HPSI and CS are not available; containment is isolated and cooled by two CACs.

6. MRIF: Small break LOCA inside containment; HPSI, CS and CACs are unavailable; containment is isolated.

7. ATWS: Automatic reactor trip and manual reactor trip mechanisms both fail. For the CCNPP IPE, potential ATWS sequences were assessed using applicable NUREG-1150 analyses. ATWS scenarios in NUREG-1150 included transients, small break LOCAs, and steam generator tube rupture events with the CS system operating (about 8% of ATWS group is for SGTRs). The NUREG-1150 ATWS analysis results for Surry were scaled to obtain CCNPP results (see Section 4.3.8.2.14 of IPE).

8. LBIO: Large break LOCA inside containment; HPSI is not available; two trains of CS are available, but fail at recirculation switchover; containment is isolated and cooled by two CACs.

9. HRSF: Station blackout with total loss of main feedwater; no SI system or CS train is available; containment has a small penetration/isolation failure; CACs are unavailable.

10. MCBO: Steam generator tube rupture; two trains of HPSI are available, but the inventory is lost outside the containment and the system fails at recirculation switchover; CS is not available; containment has small bypass due to SGTR, but is cooled by two CACs.

11. HGIP: Very small break LOCA; two trains of HPSI and CS are available; containment is isolated and cooled by two CACs.

12. HBIF: Total loss of main feedwater; HPSI is not available; two trains of CS are available, but fail at recirculation switchover; containment is isolated, but CACs are unavailable.

13. MCBF: Steam generator tube rupture; two trains of HPSI are available, but the inventory is lost outside the containment and the system fails at recirculation switchover; containment has small bypass due to SGTR, and CACs are unavailable.

14. HLBF: Small break LOCA outside containment (small V-sequence); one HPSI train is available, but the inventory is lost outside the containment and the system fails at recirculation switchover; CS is not available; containment has small bypass due to small V-sequence, and CACs are unavailable.

15. LCVF: Large break LOCA outside containment occurring in the 14" shutdown cooling return line; two HPSI trains are available, but the inventory is lost outside the containment and the system fails at recirculation switchover; CS is not available; containment has large bypass due to large V-sequence, and CACs are unavailable.

The binning of core damage sequences into PDSs seems to be reasonably well executed.

2.2.2 Containment Event Tree Development

Probabilistic modeling and quantification of severe accident progression for the probabilistically significant key plant-damage-state bins was performed using a Containment Phenomenological Event Tree (CPET). The CPET analysis starts with the 15 key plant damage states (and their frequencies) evaluated from the Level-1/Level-2 interface analysis. The result of the CPET quantification process are accident progression sequences (i.e., CPET sequences) and their estimated frequencies of occurrence. A source-term analysis, including a source-term event tree (STET), is then used to develop release categories and their frequencies. The release categories are condensed to a set of key release categories, in a manner analogous to the development of key plant damage states from PDS bins.

The main CPET developed for Calvert Cliffs has fifteen top event nodes. (In contrast, the CET developed for the NUREG-1150 analysis of Surry has 71 top events). Sub-trees were employed for a few important top event nodes in the CPET analysis. CPET quantification, in a few instances (e.g., for ATWS sequences), relies heavily on data developed for CET input quantification in the NUREG-1150 analysis of Surry. In many other cases, CPET split-fraction quantification is performed using results from plant-specific MAAP analyses (see Section 2.3.1 of this TER).

Figure 4.5.2.1 of the submittal depicts a condensed form of the CCNPP CPET used in the back-end analysis. This CPET is applicable to each KPDS (except ATWS) regardless of initiator type; i.e., CPET logic does not depend explicitly on KPDS or initiator. The split fractions used in CPET quantification do, however, depend on the specific KPDS. Thus, the same CPET is used for all KPDSs, using KPDS-specific split-fraction values. A list and brief description of the top events modeled in the CPET are discussed below.

CPET Event 1, HL: "Hot Leg Rupture or RCS Depressurization Before Vessel Melt-Through"

For transients and small LOCAs (RCS pressure > 200 psia), this top event questions whether a hot leg creep rupture occurs before vessel melt-through. Success at this branch means a low RCS pressure at VMT; failure means high RCS pressure at VMT. High pressure at VMT implies a greater potential for severe direct containment heating (DCH). In addition, depressurization could also occur as a result of appropriate operator actions. However, since the current EOPs do not direct the operator to depressurize using the PORVs, after core damage, the IPE does not model such potential operator actions. A unique feature of CCNPP (as compared to other large dry PWRs) is the low elevation of the bottom vessel head in conjunction with the fact that there are no penetrations in the bottom of the vessel. Hence, water that accumulates on the containment floor will submerge the bottom portion of the vessel, and may act to delay VMT. This may exacerbate the possibility of natural circulation-induced failure of the hot leg.

The split fractions for this top event are expressed as a failure probability (i.e., probability that RCS depressurization does not occur). Split fractions for all top events and KPDSs are presented in Table 4.6.3-A of the IPE . For top event HL, the split fractions are: 0.0 for low pressure KPDSs (LBIO, LCVF); 0.19 for high pressure KPDSs (HGIP, HBIF, HRIF, HRSF, and HLBF); 0.1 for moderate-pressure KPDSs with dry cavity (MRIO, MRIF, MCBO, and MCBF); 0.3 for moderate-pressure KPDSs with wet cavity (MBIO and MCIF); and 0.2 for KPDS ATWS. These values are comparable to the NUREG-1150 results.

CPET Event 2, DQ: "Debris Cooled and Quenched In-Vessel"

This top event addresses the possibility that the degraded core may be cooled in-vessel, preventing a VMT. Success requires that an active core cooling system becomes available or is recovered late. Only recoveries not requiring operator intervention are considered. Non-operator recovery actions apply primarily to sequences initiated by a transient or a small LOCA where HPSI has failed and the RCS pressure has remained above the LPSI shutoff pressure until hot leg creep rupture occurs. In such sequences, RCS depressurizes, and the available LPSI system can begin to inject water into the vessel, possibly preventing VMT.

The split fraction for this top event is expressed as a failure probability (i.e., that debris is not cooled in-vessel). The IPE assumed certain failure for this event; i.e., split fraction of 1.0 for all KPDSs. This is a conservative assumption.

CPET Event 3, HP: "No Hydrogen Burn Before Vessel Melt-Through"

Significant amounts of hydrogen may be generated and released into the containment atmosphere as a result of core degradation prior to vessel failure. This top event addresses the possibility that a hydrogen burn may occur before VMT, resulting in an early challenge to the structural capacity of the containment. If an early hydrogen burn does occur, and the containment remains intact, the likelihood of a severe hydrogen combustion later in the accident scenario, is reduced. Results from MAAP simulations are used to predict steam and hydrogen concentrations for the 15 KPDSs.

The split fractions for this top event depend on the steam and hydrogen concentrations estimated for various KPDSs. Following are the split fraction values (i.e., probabilities of early hydrogen combustion): 0.01 for KPDSs HRIF, HRSF, and MCIF; 0.25 for MBIO; 0.01 for HGIP and HBIF for high pressure RCS; 0.15 for LBIO; 0.1 for HGIP and HBIF for low pressure RCS; 0.03 for MRIO and MRIF; and 0. for ATWS, HLBF, MCBO, MCBF, and LCVF.

CPET Event 4, FP: "Containment Intact Up to Vessel Melt-Through"

This top event considers the possibility of containment failure prior to VMT. Early containment failure is automatic for bypass sequences and sequences associated with an isolation failure. Other potential causes of early containment failure are an early, severe hydrogen burn or other source of pressure increase.

CPET Event 5, FC: "No Cavity Event Fails Containment at Vessel Melt-Through"

This top event is supported by a Cavity Event Tree (CVET). The small volume of the cavity and the limited space between the RPV and the cavity floor, makes the RPV susceptible to "rocketing". The CVET addresses the effects of upward forces on the reactor vessel and the effect of high cavity pressures on the integrity of the biological shield. These events may potentially challenge containment integrity for high pressure vessel melt-through. For sequences involving low pressure VMT, the potential for ex-vessel steam explosions in the cavity are considered.

The following six top events describe the CVET sub-tree:

> *CVET Event 1: VD: "Vessel Melt-Through Time Delay."* A short VMT delay is assumed to be less than 30 minutes; greater than 30 minutes is a long delay. A long delay is assumed to result in a more uniform heating of the vessel bottom head and likely a larger failure area.
>
> *CVET Event 2: VF: "Vessel Failure Mode."* Depending on the heat transfer from the core debris and the submergence of the vessel bottom head, a circumferential/ring failure at the top of the molten debris, or a vessel failure at the bottom head apex, may occur. The split fraction for this event is the probability that failure occurs at the vessel bottom apex.
>
> *CVET Event 3: LF: "Melt-Through Failure Size."* This event determines the size of vessel failure, and indirectly determines the rate at which the debris and thermal energy are released into the reactor cavity. The bottom-failure hole size was <u>assumed</u> to be 30 cm, and the circumferential "ring" failure hole size was <u>assumed</u> to be 3 inches, but extending over one quarter of the vessel circumference.
>
> *CVET Event 4: VH: "Vessel Hold-Down."* Success of this event means that the vessel remains held in place; failure means that upward forces on the RPV are sufficient to shear RCS piping and launch the reactor vessel as a "rocket".
>
> *CVET Event 5: RI: "Reactor Cavity Wall Integrity."* The cavity wall failure pressure is assumed to be 300 psid. Some VMT failure modes can lead to high cavity pressures and fail the biological shield, thus potentially challenging the containment structure via ejected missiles. Success of this event means that the cavity remains intact; failure of

this event means that the cavity fails due to over-pressure at the time of VMT.

CVET Event 6: XC: "Containment Intact." This event determines the status of the containment based on the processes in the cavity at the time of VMT. Success means that the reactor cavity processes do not impair the leak tightness of the containment; failure means that the containment is no longer leak tight.

Split fraction assignments for CVET top events are discussed in Section 4.2.3.9.2 of the submittal, and details of the quantification (and justification) are provided in Appendix H of the contractor report to the licensee, which was submitted to the NRC as part of Reference [9]. In this analysis, a number of split fractions are assessed on the basis of engineering judgment. In addition a number of simplifying assumptions are made. As an example, ex-vessel steam explosions result in dynamic loads on the cavity wall, which require a dynamic capacity/fragility analysis for accurate assessment of failure probabilities. However, no such calculations are performed in the submittal. Another example pertains to the size of vessel breach in the RPV, where a number of assumptions are made. Similarly, assumptions are made regarding the cavity fragility. It is clear that the phenomena pertaining to cavity pressurization in small cavity volumes such as the Combustion Engineering (CE) design, are not well understood. The analysis undertaken in the CCNPP IPE is representative of what has been done in several other IPEs; however, it should be recognized that this treatment is still very approximate and highly uncertain. Research is currently in progress under NRC sponsorship to investigate the cavity phenomena in high pressure scenarios in Combustion Engineering plants.

The possible end states of the CVET are: (1) containment intact; (2) small containment failure from concrete debris/missiles punching through liner; and (3) large containment failure from vessel launching into/impacting the containment dome. The split fraction for CPET top event 5 provides the probability of containment failure. The following split-fraction results were obtained in the IPE: 0.097 for KPDSs HRIF, HRSF and HLBF for HL failure and a dry cavity; 0.2 for HGIP, HBIF and ATWS for HL failure and a wet cavity; 0.004 for MRIO, MRIF, MCBO and MCBF with HL failure (medium pressure) and a dry cavity; 0.012 for MBIO and MCIF with HL failure (medium pressure) and a wet cavity; 0.0 for HRIF, HRSF, MRIO, MRIF, HLBF, MCBO, MCBF, and LCVF for HL success (low pressure) and a dry cavity; and 0.0001 for HGIP, HBIF, MBIO, LBIO, MCIF, and ATWS for HL success (low pressure) and a wet cavity.

CPET Event 6, DX: "No Direct Containment Heating at Vessel Melt-Through"

For high pressure scenarios, debris may likely be dispersed into the containment after vessel breach, resulting in Direct Containment Heating (DCH) and associated containment pressurization. Success of this top event means that DCH does not occur. For the CCNPP, DCH was analyzed using the NUREG-1150 analyses, although results had to be interpreted in the IPE for use at Calvert Cliffs, since the reactor cavity configurations at Surry and Zion are somewhat different than that for Calvert Cliffs. Split fraction assignments for this event, for high pressure VMT, did not depend on whether the cavity was wet or dry.

For sequences involving low pressure VMT, debris is assumed to flow through the cavity access tunnel and spread across the containment floor, resulting in, possibly, a coolable configuration of the debris. A fuel-coolant interaction (FCI) event may, however, challenge the integrity of the cavity wall.

The split fraction expresses the probability of direct containment heating for sequences involving high and medium pressure VMT. The following split fractions assignments were made in the IPE for this top event: 0.5 for KPDSs HGIP, HBIF, HRIF, HRSF, HLBF and ATWS for high pressure sequences; and 0.25 for MBIO, MRIO, MCIF, MRIF, MCBO and MCBF for medium-pressure sequences.

CPET Event 7, HS: "No Hydrogen Burn After Vessel Failure"

This top event assesses the potential for a hydrogen burn in the containment shortly after VMT for non-DCH events. The result depends upon whether or not a hydrogen burn has previously occurred. MAAP analyses were used in the IPE to assess hydrogen and steam concentrations for evaluation of the potential for hydrogen burn.

CPET Event 8, FS: "Containment Intact After Vessel Melt-Through"

This top event considers the possibility of containment failure after VMT. The failure probability depends on the pressure in the containment just prior to VMT and the pressurization due to VMT. Potential effects of a possible hot leg creep rupture blowdown are considered, in addition to potential effects of a hydrogen burn.

Split fractions for this top event reflect the probability that the containment fails just after VMT. The split fractions developed in the IPE are numerous (depending upon various conditions), and are presented in Table 4.6.3-A of the IPE.

The most important contributor to containment failure at vessel breach is DCH, and a number of questions were posed to the licensee on the treatment of DCH in the IPE. The modelling approach of the licensee is provided as an attachment to Reference [7]. The licensee analysis of DCH is based on an extrapolation of CONTAIN results for Zion and Surry plants, which were performed in support of the NUREG-1150 analyses. The containment pressurization due to DCH was plotted as a function of fraction of core debris participating in DCH. The principal uncertainty was assumed to be in the extent of hydrogen combustion. A distribution of containment pressure was calculated for a station blackout sequence. The median of the containment pressure distribution varied from 82 to 116 psia, and the corresponding failure probabilities of the containment (obtained by convoluting with the containment fragility distribution) varied from 0.35 to 0.51. Recent research performed under NRC sponsorship has indicated that the conditional probability of containment failure for Surry and Zion is negligibly small. Hence, the IPE results appear to be conservative.

CPET Event 9, DC: "Debris Quenched and Cooled External to the Reactor Vessel"

Core debris may be coolable on the cavity floor, depending upon the configuration of debris on containment floor (which depends on the RCS pressure at the time of VMT) and upon the availability of water to quench the debris and keep it covered. The availability of water depends on the specific KPDS. The amount of debris expelled in the reactor cavity depends on the KPDS and on the outcome of top event DX.

The split fraction expresses the probability that the debris is not coolable. The IPE assesses the following split fraction values: 0.1 for KPDSs HGIP, HBIF, MBIO, LBIO, MCIF, HLBF, MCBO, and MCBF for llow pressure sequences and wet cavity; 0.75 for HBIF and HLBF for high pressure sequences, dispersed debris and not drained; 0.50 for MBIO, MCIF, MCBO, and MCBF for medium pressure sequences, dispersed debris and not drained; 0.0375 for HGIP with high pressure VMT, debris dispersed and not quenched by spray; and 1.0 for HRIF, HRSF, MRIO and MRIF. These split fractions appear to be conservative.

CPET Event 10, CS: "Core-Concrete Interaction (CCI) Source Term Scrubbed"

Failure of the preceding top event, DC, means that debris penetrates into the basemat, noncondensible gases are released from CCI, and this gas flow can strip fission products from the debris. If a substantial water pool is present above the uncooled debris, the fission products may be scrubbed and retained in the water. Scrubbing will not be effective if the water depth is too shallow.

Success of this top event means that the CCI source term is scrubbed before it reaches the containment atmosphere. The split fraction for this event reflects the probability that the source term is not scrubbed. Values assigned to split fractions in the IPE are: 0.01 for KPDSs HGIP, HBIF, MBIO, LBIO, MCIF, HLBF, MCBO, and MCBF for llow pressure VMT, wet containment, and debris in cavity; 0.25 for HBIF, MBIO, LBIO, MCIF, HLBF, MCBO, and MCBF for medium pressure sequences and wet containment; 0.0 for HGIP with high pressure VMT; and 1.0 for HRIF, HRSF, MRIO, MRIF and LCVF for dray containment and debris not cooled.

CPET Event 11, CA: "Containment Cooling Available or Recovered Late"

This event assesses the long-term availability of containment cooling. If containment cooling has been available previously, it may now not be available due to harsh environmental conditions caused by the severe-accident progression. This is the only node where the effect of harsh environmental conditions upon equipment is treated in the CCNPP back-end IPE. If containment cooling has not been previously available, it may now be available, if recovery of containment cooling is successful. Whereas the IPE admits to such a possibility, no credit for late recovery is modeled in the IPE.

The split fraction for this top event expresses the probability that containment cooling will not be available late. The split fraction assignments for this event in the IPE include: 1.0 for KPDSs HBJF, HRIF, MCIF, and MRIF with recovery of containment cooling after core damage; 0.1 for MBIO, LBIO and MRIO with hydrogen burn; 0.5 for MBIO and MRIO with a DCH event; 0.01 for MBIO, LBIO, and MRIO with no burn and no DCH; 0.02 for HGIP with burn or DCH; and 0.0 for HGIP with no burn nor DCH event.

CPET Event 12, HD: "No Large Late Hydrogen Burn Fails Containment"

A hydrogen burn may occur late in an accident sequence. The likelihood of such an event depends on the occurrence of earlier hydrogen burns as well as on the magnitude of hydrogen generated as a result of CCI. Two types of late hydrogen burns may occur: small burns which do not challenge containment integrity, or a large burn which can fail containment. Failure of this top event means that a large late hydrogen burn occurs and fails the containment. Probability of failure for a given peak pressure is assessed from the containment fragility analysis.

The split fraction for this event expresses the probability that containment failure associated with a late hydrogen burn occurs. Split fraction values developed for this event in the IPE are numerous, and can be found in Table 4.6.3-A of the IPE.

CPET Event 13, FD: "Containment Intact Long Term"

Long-term challenges to containment capacity, resulting from slow pressure increases, can result from CCI-generated gases and other over-pressurization mechanisms. This event is concerned with whether or not such challenges actually fail the containment. If containment cooling via sprays or air coolers is not available, then overpressure failure is a certain eventuality. Hence, this event is only applicable to those cases where containment cooling is still functioning late. The source terms associated with late containment failure are not expected to be as severe as for early failures, since much of the fission products will have been retained in containment.

The split fraction for this event measures the probability that the containment fails long term. Values for split fractions developed in the IPE are: 0.001 for KPDS HGIP with debris not cooled; 0.001 for MBIO with debris not cooled; 0.0001 for LBIO with debris not cooled; and 0.001 for MRIO with debris not cooled.

CPET Event 14, LD: "No Gross Containment Failure"

This question is concerned with the size of the leak associated with containment failure, which effects the characteristics of the source term of the applicable release category. For over-pressure failure modes, the leak size is inferred from the containment structural analysis; otherwise, the size is determined from the CVET or the event tree for leak size from V sequences. A large leak will result in a rapid release of fission products, whereas a small leak results in an extended, attenuated release.

The split fraction for this event expresses the probability that a large, gross containment failure occurs. Numerous split fraction values are developed in the IPE for this top event and are presented in Table 4.6.3-A.

CPET Event 15, BM: "No Leak From Basemat Melt-Through"

The last top event in the CCNPP CPET is concerned with whether or not basemat melt-through occurs for those sequences that have resulted in containment survival up to this point (containment cooling maintained), but in a debris configuration that has not been cooled. If the ablation of the concrete basemat is arrested before melt-through occurs, then this event is successful.

The split fraction for this event expresses the probability that basemat melt-through eventually occurs. The following split-fraction values are developed in the IPE: 0.9 for KPDS LBIO with debris not cooled; 0.01 for HGIP with debris not cooled; and 0.5 for MBIO and MRIO where the debris is not cooled.

In general, the CPET analysis for Calvert Cliffs considers all phenomena of interest to severe-accident progression at CCNP. As already noted, some aspects of the analysis are similar to the NUREG-1150 CET analysis for Surry. The CPET model for Calvert Cliffs, which involves 15 top event nodes, is not nearly as detailed as the Surry model, which involves 71 top events. Two particular phenomena, namely, processes that occur at vessel breach including DCH, and cavity pressurization at vessel breach, have been treated in some detail.

2.2.3 Containment Failure Modes and Timing

The containment failure modes and timing analysis for the CCNPP back-end IPE is supported by a plant-specific containment fragility analysis of the pressure resistance (as a function of temperature) of the steel-lined pre-stressed concrete containment building. This containment structural evaluation is summarized in Section 4.4 of the CCNPP IPE. The analysis considers 12 failure modes/locations; 5 of these are associated with gross containment failure, and 7 are associated with leakage. In addition, the 12 modes are comprised of 6 structural failure modes and 6 penetration failures. The fragility analyses are performed for three temperatures, namely, 400°F, 600°F and 800°F. Correlation in strength and modeling uncertainties were accounted for in the analysis when combining the individual failure modes into a composite containment fragility curve.

The fragility parameters for pressure capacity of seven controlling modes of containment failure at CCNP are listed in Table 3 of this report. In this table, parameter P is the median pressure capacity against the designated failure mode; β_M is the logarithmic standard deviation associated with modeling uncertainty; β_S is the logarithmic standard deviation associated with strength variability; and β is the overall logarithmic standard deviation associated with the designated failure-mode capacity. The values in this table for overall logarithmic standard deviation are not dissimilar to values obtained in other studies.

Table 3: Fragility Parameters for Pressure Capacity Associated with Controlling Modes of Containment Failure (400°F)

Failure Mode	Mode No.	Failure Type	P (psig)	β_M	β_S	β	95% Conf. (psig)
Liner Tear around Personnel Airlock	1	Leak	132	0.22	0.05	0.23	90
Basemat Shear	2	Break	143	0.22	0.08	0.23	98
Cylinder Hoop Membrane	3	Break	158	0.13	0.05	0.14	125
Wall-Basemat Junction Shear	4	Break	164	0.20	0.11	0.23	112
Cylinder Meridional Membrane	5	Break	180	0.14	0.05	0.15	141
Dome Membrane	6	Break	262	0.13	0.05	0.14	208
Personnel Airlock Door Buckling	7	Leak	180	---	---	0.24	121

All load conditions considered in the analyses were for quasi-static pressurization under various temperature conditions; dynamic effects were not considered. The containment structural failure assessment is, therefore, not relevant to dynamic loads resulting from energetic phenomena (e.g., fuel coolant interaction).

The failure-mode fragilities are combined into a single, composite containment fragility curve, accounting for correlations among variabilities in capacities for the various modes. The composite fragility curves thus obtained have the following median pressure capacities: 131 psia (117 psig) for 400°F; 126 psia (112 psig) for 600°F; and 124 psia (110 psig) for 800°F. These median capacities are considered to be reasonable.

The containment analysis evaluates, as a function of pressure and temperature, the fraction of containment failures that result in rapid depressurization (containment rupture) and failures that result in slow depressurization (leak-before-break). Containment pressurization is judged to stabilize when a leak size equivalent to a 3" to 4"-diameter hole (7 to 13 sq. in.) is reached. Figure 4.4.2-A of the IPE presents a plot of expected leak area for the CCNPP containment, as a function of pressure and temperature conditions; this plot is represented in the MAAP input files for analysis of severe-accident progression. The basis for derivation of Figure 4.4.2-A was provided by the licensee in response to NRC team RAIs [7], and the calculations for break (and leak) size indicate that the results are appropriate . Fragility curves for individual failure modes are plotted in Figures 4.4.1-A through 4.4.1-I.

In general, the CCNP containment fragility analyses documented in Reference [7], are well executed.

2.2.4 Containment Isolation Failure

Containment isolation and bypass failures were analyzed in the IPE, although the Level-2 documentation does not contain many details on these analyses. Both small and large isolation and bypass failures were considered. Containment isolation failure was analyzed using the front-end event trees. The calculated frequency of failure of containment isolation is 1.11×10^{-5} per reactor year (conditional probability of 4.9%). Based on PDS definitions there are no large containment isolation failures (other than the large bypasses); the only PDS with a small containment isolation failure is KPDS HRSF, which involves a small penetration (<4" in diameter) not being isolated.

Both SGTRs and V-sequences were analyzed. The frequencies of small and large bypass categories evaluated in the CCNPP back-end IPE are summarized in Table 4 of this TER. Small bypasses contribute 3.1% to the total CDF; large bypasses contribute 0.16%.

The calculated conditional probabilities for bypass and isolation failure contributions appear to be reasonable and are consistent with other IPEs.

2.2.5 System/Human Response

The licensee chose not to model operator actions in the back-end analyses. The rationale for not modelling the operator actions in the submittal [7] is that "the existing EOPs are not intended to apply to conditions after core damage, and therefore should not automatically assumed to be followed, particularly if they had not been followed before core damage started". Hence, the licensee concludes that "most actions identified as potential recovery actions (e.g., depressurization by opening a PORV when the core exit thermocouples indicate excessive temperatures) should be taken before core damage, and if credit is taken, it should be taken in the level 1 model". In addition, the licensee also stated that they decided not to take credit for actions which are not proceduralized and reviewed by operations personnel.

2.2.6 Radionuclide Release Categories and Characterization

The CPET analysis in the submittal resulted in 444 CPET sequences for each of the 15 KPDSs that served as initial conditions to the analysis. These 444 CPETs are assigned to source-term release categories based on a Source Term Event Tree (STET) analysis. The STET analysis is analogous to the PDS binning of Level-1 sequences in the interface analysis. The STET effectively bins the CPET sequences into a set of release categories. The result of the STET analysis is a set of 36 release categories. The top events developed in the STET are described briefly below:

STET Event 1: "Debris Quenched and Cooled In-Vessel"

This node in the STET determines on whether of not the core debris is cooled in-vessel. This STET top event is based directly on CPET top event DQ.

STET Event 2: "Containment Failure Time"

The time of containment failure (relative to the time of core damage) is used to define the start of release for a source term. The release is categorized as early, if (1) the containment is failed, or if the containment is unisolated or bypassed, at the time of core damage, or if (2) the containment fails anytime prior to about two hours after VMT. It was judged in the IPE that there would be a very limited time for release mitigation by natural processes (e.g., aerosol agglomeration and deposition) for early containment failure. A release is categorized as late if the containment is isolated and remains intact after vessel melt-through, but failure occurs in the long-term (~24 hours after VMT).

Therefore, four possible modes of containment failure time are possible: (1) containment remains isolated and intact; (2) delayed (late) containment failure several hours after VMT; (3) early containment failure, associated with initial isolation failure or CPET top events FP, FC or FS; and (4) bypass containment failure (V-sequence).

STET Event 3: "Containment Failure Mode"

This node distinguishes between leak-before-break failure modes and gross containment failures. In addition to large breaks, gross failure is associated with large penetration failures and large bypass sequences (V sequences). Gross failure results in short-duration source terms, the magnitudes of which depend, in part, on the timing of containment failure.

STET Event 4: "Configuration of Debris In Cavity"

The configuration of debris on the containment floor, including the depth of water pool on the containment floor, impacts the extent of CCI and the degree of fission-product scrubbing. This top event thus distinguishes whether the debris is quenched and cooled, or uncooled, and whether the vaporization source term is scrubbed or unscrubbed.

STET Event 5: "RCS Pressure at Early Vessel Melt-Through"

The last STET top event distinguishes the RCS pressure at VMT for early containment failure modes. This STET node is determined by CPET top event HL.

The KPDS frequencies and CPET split fractions were used in the IPE to quantify the frequencies of CPET accident-progression sequences for each KPDS. The STET was applied to the CPET sequences to obtain frequencies for each of the 36 source terms/release category bins resulting from the STET analysis. These 36 release categories were further condensed to six (6) key release categories and their associated frequencies. This condensation analysis is described in Sections 4.7.4 to 4.7.6 of the IPE. The six key release categories and their frequencies are presented in Table 4.

Table 4: Designation and Frequency of Key Release Categories

Key Release Category	Frequency per reactor year (Conditional probability)
I. Intact Containment	1.1×10^{-4} (48.6%)
II. Late Containment Failure	8.9×10^{-5} (40%)
III. Small Early Containment Failure	1.4×10^{-5} (6.2%)
IV. Large Early Containment Failure	5.2×10^{-6} (2.3%)
V. Small Containment Bypass	7.0×10^{-6} (3.1%)
VI. Large Containment Bypass	3.5×10^{-8} (0.01%)

These key release categories appear to be reasonably defined and provide a useful, concise summary of the Level-2 analysis. The 36 release categories and their frequencies, however, are valuable in presenting a somewhat more refined summary of the analysis. Radiological source terms are evaluated in the IPE only for the key release categories. The source terms are based on MAAP analyses of the top CPET sequence in that key release category.

No source terms were reported for the key release category I (intact containment). For the next release category II (late containment failure), eleven sequences have a frequency of greater than 10^{-6} per reactor year. The top sequence is from KPDS HRIF, and contributes to 28% of the frequency of the release category. The source terms for this sequence is used to determine the source term for this release category, and are listed in Table 5.

The next release category, small early containment failure, is dominated by a sequence from KPDS HRSF involving a failure to isolate containment. The source terms for this sequence are used to define the source terms for the release category. For the release category IV involving large early containment failure, results from the NUREG-1150 analyses for the Surry plant (for early containment failure for a transient) were used. For the two containment bypass categories, source terms were calculated using the MAAP code. The source terms for all release categories are listed in Table 5. However, two points should be noted regarding the results for the calculated source terms: first, the release fraction of noble gases for late containment failure and small bypass release categories is less than one. This is an artifact of the time period (after containment failure) over which the MAAP simulations were performed. In the long term, the entire inventory of noble gases in the core must be released to the environment. The second point pertains to the release fractions of cesium and iodine calculated for large bypass sequences, which is larger than results calculated using the MELCOR and STCP codes. However, it should be noted that the release frequency of large bypass sequences is small in the IPE submittal.

Generic Letter 88-20 states that: "any functional sequence that has a core damage frequency greater than or equal to 10^{-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, and any bypass sequence that has a core damage frequency greater than or equal to 10^{-7} per reactor year" should be reported by the IPEs. The licensee has addressed this request, and the submittal provides a list of all the reportable sequences in Table 4.7.5 of the submittal, and the corresponding source terms in Table 4.7.6 of the submittal.

Table 5 Source Terms for Various Release Categories

Key Release Category	Noble Gas	I	Cs	Te	Sr
II. Late Containment Failure	0.2	9.5×10^{-4}	7.6×10^{-4}	7×10^{-3}	1.5×10^{-6}
III. Small Early Containment Failure	1.0	0.13	0.13	0.58	4.8×10^{-4}
IV. Large Early Containment Failure	1.0	0.19	0.20	0.023	5×10^{-4}
V. Small Containment Bypass	0.7	2.8×10^{-3}	2.7×10^{-3}	2.8×10^{-5}	3.4×10^{-6}
VI. Large Containment Bypass	1.0	0.69	0.69	0.032	1×10^{-3}

2.3 Quantitative Assessment of Accident Progression and Containment Behavior

2.3.1 Severe Accident Progression

Section 4.2 of the CCNPP IPE describes nine plant-specific analyses that support the CPET analysis and quantification for severe-accident progression. These analyses were performed to provide input data to MAAP and to address physical processes and uncertainties in CPET branch points. These plant-specific analyses are summarized below:

1. Analysis A: Stress-Strength Interference Analysis of Containment Failure Probability. This analysis used the containment fragility curves and pressure/temperature conditions predicted for various accident progression, in order to develop probabilities of containment failure for CPET sequences. This analysis also evaluates the mode/size of containment breach.

2. Analysis B: RCS Failure Modes at High Pressure. This analysis evaluates the potential for the following three RCS failure modes for high core pressure sequences: (1) creep rupture of RCS hot leg or pressurizer surge line; (2) creep rupture of the steam generator tubes; and (3) VMT by molten debris attack on the vessel bottom head.

3. Analysis C: Direct Containment Heating. This analysis examines potential containment conditions resulting from direct containment heating (DCH) associated with high pressure melt ejection and related debris transport and heat transfer phenomena. The analysis estimates probability distributions for containment pressure loads associated with DCH progressions and various hydrogen burn scenarios.

4. Analysis D: Containment Response to Combustion of Hydrogen. This analysis assesses containment pressures resulting from global and continuous discharge (local) burning of hydrogen. The analysis accounts for Zr clad oxidation and containment steam concentration.

5. Analysis E: Containment Failure Due to Late Hydrogen Burning. This analysis assesses the potential and containment response for late hydrogen burns that may occur.

6. Analysis F: Time Delay Between Core Slump and Vessel Melt-Through. This analysis develops a best estimate of the VMT time delay based on MAAP results and VMT-delay analyses conducted for IDCOR 15.2A. Results are compared with results from similar analyses for plants Surry and Sequoyah. For low RCS pressures, the CCNPP VMT delay times are developed to be substantially shorter than those for Surry and Sequoyah. The following VMT delay times are used in the CCNPP IPE: (a) 20 minutes for RCS pressures greater than 1,300 psia; (b) 30 minutes for RCS pressures between 200 and 1,300 psia; and (c) 40 minutes for RCS pressures less than 200 psia.

7. Analysis G: Reactor Cavity Pressure at Vessel Melt-Through. This analysis estimates a range of reactor cavity pressures for VMT. MAAP is used to establish lower-bound pressures; upper-bound pressure values, that include debris-entrainment effects, are evaluated based on estimates developed by Gabor, Kenton and Associates. The cavity pressure estimates are applied for both bottom-head vessel melt-through and ring/circumferential vessel breach.

8. Analysis H: High Pressure Phenomena in the Reactor Cavity at Vessel Breach. This analysis is equivalent to the cavity event tree (CVET) analysis discussed in Section 2.2.2 of this TER. The objective of the analysis is to determine the potential for containment liner breach or dome-impact failure as a direct consequence of catastrophic phenomena in the reactor cavity.

9. Analysis I: Containment Overpressure Failure Analysis. This analysis is equivalent to the containment fragility study described in Section 4.4 of the IPE and reviewed in Section 2.2.3 above.

MAAP Version 3.0B, Revision 19.01 was the principal tool used to analyze postulated severe accidents at Calvert Cliffs (for the plant-specific analyses). The MAAP input files are not provided as a part of the submittal, but the input file and the supporting calculations were developed by BG&E and are documented in a BG&E (internal) document. MAAP analyses were conducted for postulated representative sequence conditions associated with each KPDS (except ATWS), in order to quantify KPDS-specific split fractions for use in quantifying the CPET. Various conditions were introduced into the analyses, but only a single set of split fractions was generated for each CPET top event and each KPDS (i.e., uncertainties in split-fraction values were not developed). MAAP analyses were also conducted for postulated representative sequence conditions associated with each key release category (5 total), in order to develop estimates of radiological source terms for KRCs. In addition, MAAP analyses were conducted for various aspects of the 9 supporting analyses described above.

Most of the assessment of severe-accident progressions for the CCNPP IPE are derived from CCNPP-specific MAAP analyses. A few aspects of the analysis are derived from the Surry NUREG-1150 study. In some cases, due to differences in plant and containment characteristics, it is questionable that the results for Surry are applicable to Calvert Cliffs. The usage of these results in the CCNPP IPE is thought to be generally appropriate. Other instances of quantification in the CPET analysis are based solely on engineering judgment (of the analysis team or others).

The quantification of accident progression in the CCNPP CPET analysis appears to be reasonable, logical and generally acceptable. However, the extent of sensitivity analyses performed in the Calvert Cliffs IPE submittal is very limited. It would have been valuable for the licensee to assess the risk sensitivity to a number of parameters/assumptions, in order to obtain additional important insights.

2.3.2 Dominant Contributors to Containment Failure

Table 6 of this review shows a comparison of the conditional probabilities of the various containment failure modes of the CCNP IPE submittal with the Zion and Surry (NUREG-1150) results [2]. All comparisons are made for internal initiating events only.

The CCNP core damage frequency for internal events is approximately one order of magnitude larger than the CDF calculated by NUREG-1150 for Zion [5]. The conditional probability of early containment failure in the CCNP IPE (due to overpressurization and α-mode failure) is approximately 3.6%, which is approximately a factor of five larger than the results calculated by the NUREG-1150 analyses. The differences in the results can be attributed to the somewhat conservative treatment of the DCH issue by the licensee, and the identification of vessel rocketing as an important mode of early containment failure.

The conditional probability of late containment failure (40%) calculated by the CCNP IPE is also larger than that calculated for the NUREG-1150 analyses. The assumption of assured late containment failure if containment cooling is not available in the accident scenario, is an important reason for the calculation of a rather high conditional probability of late containment failure. In addition, the calculated conditional probability of isolation failure (which, the licensee groups together with early containment failure) is also larger in the CCNP IPE, in comparison to other IPE submittals.

Table 6 Containment Failure as a Percentage of Total CDF: Comparison with Other PRA Studies

Containment Failure Mode	CCNP IPE	Surry NUREG-1150	Zion+ NUREG-1150
Early Failure	3.6	0.7	0.5
Late Failure	40	5.9	24.0
Bypass (V)	0.01	7.6	0.2
Bypass (SGTR)	3.1	4.6	0.3
Isolation Failure	4.9	NA++	1.0
Intact	48.6	81.2	73.0
Core Damage Frequency, yr^{-1}	2.4×10^{-4}	4.1×10^{-5}	6.2×10^{-5}

++ Included as a part of Early Containment Failure
+ After Charging Pump Modification

2.3.3 Characterization of Containment Performance

The progression of various accident scenarios were modelled and quantified using an event tree. The results of the containment analyses for the important release modes, are discussed below.

Release Categories III and IV: Early Containment Failure and Isolation Failure

This release category has a frequency of 1.92×10^{-5} per reactor year, and accounts for 8.6% of the total CDF. Isolation failure is the biggest contributor to this release category (56%), followed by vessel rocketing (16%), DCH (12%), and cavity pressurization accompanying vessel breach (8%). The biggest contributors, by initiating event to the early releases are station blackout damage state HRSF, and small break LOCA damage state HGIP.

Release Category II: Late Containment Failure

This release category has a frequency of 8.9×10^{-5} per reactor year, and accounts for 40% of the total CDF. Containment overpressure failure is the biggest contributor to containment failure The biggest contributor is the station blackout damage state HRIF.

2.3.4 Impact on Equipment Behavior

Equipment survivability in a severe accident environment in the Calvert Cliffs plant is discussed in the submittal [1] and in response to the NRC team questions [7]. The licensee considers the survivability of the following equipment: sprays, fan coolers, the instrumentation and control systems.

Sprays and Fan Coolers: Equipment vulnerability to conditions caused by hydrogen combustion was considered in the CPET. The only equipment which impact the results, and are taken credit for in the back-end analyses, are the air coolers and the containment sprays. The CET top event CA considers the availability of the fan coolers and the sprays for long term heat removal. For the containment spray system, the containment spray headers, several manual valves, a check valve and an air-operated valve are located inside the containment. The licensee concludes that these components are not vulnerable to hydrogen combustion, and the conditional probability of failure of the spray system, was assigned a value of 0.02. Failure of air coolers is more probable because air cooler power and control cables, fans and motors are located inside the containment. A split fraction of 0.1 was assigned to the failure of air coolers due to combustion.

The operability of the sprays and fan coolers after energetic non-combustion events, such as blowdown following vessel breach and DCH, was treated in a similar manner. However, results show a negligible impact of such events upon the functioning of equipment.

Three CPET accident sequences indicate failure of containment cooling systems after a hydrogen combustion, and thus leading to containment failure. These sequences correspond to a conditional probability of containment failure of approximately 2.2% of the CDF.

Instrumentation Sensors: The licensee does not credit the functioning of instrumentation in the containment after core damage.

PORV Valves: Once again, the licensee does not credit the availability of PORVs in the submittal after core damage.

Recirculation and Spray Sump Plugging: The emergency sump is protected from the containment floor by a 16 inch curb. Even if the entire core inventory relocated to the containment floor, debris cannot spread to the sump.

Mechanical and Electrical Penetrations: The containment structural analysis considered the impact of elevated temperatures (up to 800°F) upon mechanical and electrical penetrations, and found that failure of these penetrations were not a threat to containment integrity. At all temperature levels, liner tearing at the personnel airlock was the dominant containment failure mode.

2.4 Reducing the Probability of Core Damage and Fission Product Releases

2.4.1 Definition of Vulnerability

The licensee does not provide an explicit definition used to identify a "vulnerability" in the back-end IPE. No vulnerability assessment was documented for the back-end analysis.

2.4.2 Plant Modifications

Several plant improvements were identified and implemented in response to a plant-specific vulnerability assessment. However, these plant improvements related to insights that were derived from the front-end analysis. The insights derived from the Level-2 IPE did not result in implementation of plant improvements, but did produce some observations concerning accident management considerations and evaluation of phenomenological uncertainties. These are summarized in the IPE submittal (Section 4.8) as follows:

Accident Management Considerations

1. Late containment failure is mitigated by late recovery of containment heat removal. Alternatively, accident management actions to provide alternate water injection into the containment and/or to open a small containment vent, late in an accident sequence, may avoid or delay a late over-pressure failure mode. There is no existing containment vent at the Calvert Cliffs nuclear plan.

2. Given a station blackout, the probability of failing to isolate a containment penetration is about 0.25 %. Several hours should be available for plant operators to manually close/isolate any failed penetrations.

3. It may be feasible and desirable to depressurize the reactor coolant system (RCS) by opening the pressurizer PORVs, if indication of extended core uncovery is received. For SGTR sequences, RCS depressurization may allow isolation of the faulted steam generator. For early large containment failure sequences, this action may avert early containment failure due to vessel rocketing. This accident management consideration would also have benefits for other sequences with a high pressure vessel melt-through.

4. Combined RCS depressurization and increased RWT injection is a candidate strategy for mitigation of severe accidents at CCNPP. RWT injection submerges the bottom 5 to 6 feet of the reactor vessel. Before vessel melt-through, the molten debris in-vessel will build up to about the same level. Even though the bottom portion of the vessel is cooled, the vessel wall above the debris is postulated to heat up, likely leading to a delayed vessel failure under high RCS pressure conditions (because of the stresses from the high pressure acting in conjunction with vessel heat-up). If the water level in the containment can be raised about 5' more, by injecting twice the RWT volume, then vessel melt-through may be averted altogether if the RCS can also be depressurized.

Insights from Evaluation of Phenomenological Uncertainties; Possible Situations

1. Unlike many other PWR designs, there are no penetrations through the bottom of the reactor vessel at CCNPP. The molten debris must penetrate the entire 4-3/8" thick lower vessel head before the vessel is breached. Thus, melt-through is delayed, with resulting possible effects on accident progression:

    - For high pressure sequences, the likelihood of a beneficial hot-leg creep failure is higher because the hot legs are at a higher temperature longer.

    - For small LOCAs (including stuck open PORV), the RCS has time to depressurize after the water in the vessel boils off.

    - A larger area of eventual melt-through is likely to result.

    - Because the entire bottom head of the vessel is at higher temperature longer, the possibility of a creep failure is greater.

    - The average debris temperature at vessel failure will be higher.

    - A "ring" (or circumferential) melt-through may occur, as opposed to a bottom head melt-through, because heat flux may become highest at the top outside edge of the heated debris pool.

    - For a ring failure mode in conjunction with a high vessel pressure, the vessel breach area may propagate in an unstable manner around the circumference of the bottom head, before the vessel is depressurized by the blowdown.

    - For a ring failure in conjunction with high vessel pressure, an upward force on the vessel may be developed that exceeds the resistance of the heated hot and cold legs. The vessel may thus be launched through the missile shield and possibly the containment dome.

2. As noted above, complete RWT injection results in submergence of the bottom 5 feet of the reactor vessel. This level of submergence may not be adequate to avert vessel melt-through, but rather may exacerbate the (delayed) ring vessel failure mode for high pressure sequences, resulting in an even hotter bulk debris (due to the melt-through delay) than without water cooling of the vessel. Hence, upon vessel breach, direct containment heating (DCH) and high pressure cavity effects may be more severe.

3. High cavity pressures for high pressure melt ejections may fail the biological shield or tear out penetrations, either of which may result in gross overall containment breach.

4. It was judged from the IPE that the best accident management strategy to mitigate phenomenological consequences of possible dominant severe accidents is not obvious. The following observations were, however, drawn:

- RCS depressurization before vessel melt-through is desirable for all high pressure sequences.

- RWT injection before vessel melt-through may not necessarily be a better strategy for consequence mitigation than a low pressure melt-through into a dry cavity with subsequent RWT injection.

- An accident-mitigation strategy that depressurizes the RCS before vessel melt-through and keeps the RWT out of the cavity until after vessel melt-through is cited in the IPE as offering the least uncertainty with respect to potential upper-bound consequences of severe accidents at CCNPP.

The IPE states that the foregoing insights will be considered in the development of CCNPP Severe Accident Management guidelines. However, it should be pointed out here that many of the accident management considerations and insights obtained, are not based on realistic analyses. As an example, consider the fourth accident management consideration listed in page 28 of this TER. The licensee believes that RCS depressurization in conjunction with injecting approximately twice the volume of the RWT into the cavity, is necessary for preventing vessel breach. However, no calculations are shown in the submittal to prove this case. Other calculations have shown that flooding of water to levels above the height of the debris is sufficient to prevent vessel breach. In addition, the licensee believes that complete vessel melt-through due to melt attack, is necessary for vessel failure. Partial melting of the vessel wall is sufficient to cause vessel breach for most accident sequences. In summary, some of the insights gained from the analysis and accident management considerations are not fully supported by analyses, and have to be thoroughly reviewed before putting to practice. In addition, it appears that no plant improvements will result from the Level-2 analysis.

2.5 Responses to CPI Program Recommendations

Generic Letter 88-20, Supplement Numbers 1 and 3 [7-8] identified specific Containment Performance Improvements (CPIs) to reduce the vulnerability of containments to severe accident challenges. For PWRs with large dry containments, it is requested that the licensee evaluate the IPE results for containment and equipment vulnerabilities to hydrogen combustion (local and global), and determine the need for procedural and/or hardware improvements. The licensee has analyzed the vulnerability of the CCNP containment to the potential for hydrogen pocketing [7]. The licensee noted that all compartments in the lower containment which contain the primary system components, are all open at the top, communicating with the upper containment free volume. Thus, hydrogen released from the primary can rise to the upper compartment. The only exception is the pressurizer cubicle, where there is a region about 5.5 feet in height (cross-section of 12.5 feet x 15 feet) above the pressurizer, which is bound by concrete walls and a ceiling. This region communicates to the upper compartment of the containment through an opening 2 feet x 5 feet in cross-section. Inside the cubicle, above the pressurizer, there are PORVs, associated block valves, piping, instrumentation, and electrical junction boxes. It is possible for hydrogen to accumulate in the top 5.5 feet of the cubicle (approximately 1000 ft^3 in volume), only if a LOCA occurs in the piping inside the cubicle. However, since the length of the RCS piping inside the cubicle is small, and the possibility of a LOCA in the piping is rather remote. Ignition sources exist in the cubicle, and the licensee believes that the ignition sources will ignite the hydrogen before an explosive accumulation occurs. Under the remote possibility of a Deflagration-to-Detonation Transition (DDT) in the cubicle, a shock wave will propagate to the metal wall of the cubicle. The sheet metal wall is designed to fail at 5 psid. The failure of the wall is expected to prevent the reminder of structure from failure. Since the sheet metal wall faces the inner portion of the upper containment, the licensee believes that blowout of the wall will not lead to projectiles that can fail the containment.

Global hydrogen combustion was considered in the IPE submittal, and the conditional probability of containment failure at vessel breach due to hydrogen combustion was assessed to be in the range of 0.0015 to 0.033 (for different KPDSs). However, the conditional probability of containment failure due to detonation was assessed to be of the order of 10^{-4}.

Equipment vulnerabilities to conditions caused by hydrogen combustion were considered in the submittal, and were discussed in Section 2.3.4 of this report.

3. CONTRACTOR OBSERVATIONS AND CONCLUSIONS

The back-end portion of the Calvert Cliffs IPE submittal provides a substantial amount of information in regard to the subject areas identified in Generic Letter 88-20 and NUREG-1335. The analysis is, in general, logically prepared, well executed, and clear and concisely presented.

The important points of our technical evaluation of the CCNPP IPE back-end analyses are summarized below:

- The Back-End portion of the IPE supplies a substantial amount of information with regards to the subject areas identified in Generic Letter 88-20, and NUREG-1335. For the most part, the separate models used in the CCNP IPE Back-End analysis are technically sound. Most sections of the IPE are well written and concise.

- The licensee has addressed all phenomena of importance to severe accident phenomenology in PWRs. The licensee has also recognized the importance containment failure modes relevant to CE plants, such as vessel rocketing and cavity failure at vessel breach.

- The treatment of phenomenological issues in the CPET appears reasonable, and a good balance is struck between use of plant-specific results, generic results, results from NUREG-1150 analyses, and use of engineering judgment.

- The submittal has addressed the recommendations of the CPI program (GL 88-20, Supplements 1 and 2).

- The conditional probability of early containment failure at CCNP (due to overpressurization, vessel rocketing, etc.) is 3.6%, and is approximately a factor of five larger than that calculated in the NUREG-1150 analyses for Zion or Surry. The differences in results can be attributed to a rather conservative treatment of DCH in the CCNP IPE submittal. Additionally, the configuration of cavity at CCNP leads to the possibility of containment failure due to rocketing, and due to cavity failure at vessel breach. Identification of these failure modes have contributed to the calculation of a larger value for the conditional probability of early containment failure.

- The conditional probability of late containment failure at CCNP (due to overpressurization and combustion of non-condensible gases generated by MCCI) is 40%, and is larger than that calculated for Zion (24%) or Surry (5.9%). It is assumed in the submittal that the unavailability or failure of containment heat removal after core damage, always leads to late containment failure, if containment has not failed earlier. No credit is given for the recovery of AC power.

In conclusion, the Back-End portion of the IPE is well performed and provides information with regards to the subject areas identified in Generic Letter 88-20, and NUREG-1335. Licensee involvement in the back-end analyses was limited, and the analyses appear to have been performed with the help of outside contractors. The event tree developed by the licensee for containment analyses includes all the phenomena of importance to severe accident progression in PWRs. The licensce's process for the evaluation of containment failure probabilities and failure modes is consistent with the intent of Generic Letter 88-20, Appendix I. The dominant contributors to containment failure are consistent with the insights obtained from the NUREG-1150 analyses for the Surry plant. Additional modes of containment failure specific to the CE design have been identified. The licensee has considered the failure of containment isolation system and containment bypass. Failure of electrical and mechanical penetrations at elevated temperatures were considered and ruled out. The submittal has addressed the recommendations of the CPI program, requested as part of the GL 88-20, Supplements 1 and 2. No "vulnerabilities" have been identified. However, a number of insights and suggestions for accident management have been developed.

4. REFERENCES

1. "Calvert Cliffs Nuclear Power Plant Individual Plant Examination Summary Report," Baltimore Gas and Electric Company, December, 1993.

2. USNRC, "Individual Plant Examination: Submittal Guidance," NUREG-1335, August, 1989.

3. USNRC, "Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR §50.54(f)," Generic Letter 88-20, November 23, 1988.

4. "Severe Accident Risk: An Assessment of Five U.S. Nuclear Power Plants," NUREG-1150 (1990).

5. "Evaluation of Severe Accident Risks: Surry Unit 1", U. S. Nuclear Regulatory Commission, NUREG/CR-4551, Vol. 3, Part 1 (June 1990).

6. "Evaluation of Severe Accident Risks: Zion Unit 1", U. S. Nuclear Regulatory Commission, NUREG/CR-4551, Vol. 7, Part 1 (June 1990).

7. NRC Letter to All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities, "Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR §50.54(f)," Generic Letter 88-20, Supplement No. 1, dated August 29, 1989.

8. NRC Letter to All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities, "Completion of Containment Performance Improvement Program and Forwarding of Insights for Use in the Individual Plant Examination for Severe Accident Vulnerabilities - Generic Letter No. 88-20 Supplement No. 3 - 10 CFR §50.54(f)," Generic Letter 88-20, Supplement No. 3, dated July 6, 1990.

9. "Response to the Request for Additional Information for the Individual Plant Examination Program Calvert Cliffs Nuclear Power Plant Units Nos. 1 and 2," Baltimore Gas and Electric Company, September 12,1995.

APPENDIX A

IPE EVALUATION AND DATA SUMMARY SHEET

BWR Back-End Facts

Plant Name

Calvert Cliffs Nuclear Power Plant

Containment Type

Large Dry

Unique Containment Features

- Pre-stressed, post-tensioned containment design.
- The reactor vessel bottom head is located at about the same level as the containment floor, thus the lower vessel head is submerged (up to about a 5 feet depth) in accident sequences involving the injection of RWT into the containment.
- There are no penetrations in the bottom of the reactor vessel. The biological shield serves as the wall of the reactor cavity; there is a 30 inch square opening in the cavity wall that permits debris spreading on the containment floor.

Unique Vessel Features

As noted above, the vessel bottom head is located at the same level as the containment floor.

Number of Plant Damage States

166 modeled;
56 quantified as having non-zero frequency;
15 key plant damage states are condensed from the larger set of PDSs.

Containment Failure Pressure

131 psia at 400°F;
126 psia at 600°F;
124 psia at 800°F.

Additional Radionuclide Transport and Retention Structures

Surface area of structural elements and large equipment provide a means for decontamination and retention of fission products. Fission product scrubbing takes place in recirculation mode of containment spray and safety injection. ECCS pumps, sump drainage, and spent fuel pool are all in the Auxiliary Building, which may be involved in fission product retention in some accident progressions. However, no credit taken for additional retention structures in the submittal.

Conditional Probability That the Containment Is Not Isolated

4.9% for small isolation failures.

Important Insights, Including Unique Safety Features

See Executive Summary and Section 3 of this review.

Implemented Plant Improvements

No Level-2 related plant improvements were noted; some potential considerations for Accident Management guidelines were generated.

C-Matrix

(14×31) matrix for 14 key plant damage states and 31 release categories provided in Tables 4.7.3A and 4.7.3B of the submittal.

CONCORD ASSOCIATES, INC.

Systems Performance Engineers

CA/TR 96-019-34

CALVERT CLIFFS NUCLEAR PLANT, UNITS 1 AND 2

TECHNICAL EVALUATION REPORT ON THE IPE SUBMITTAL HUMAN RELIABILITY ANALYSIS

FINAL REPORT

by

P.J. Swanson

Prepared for

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Division of Systems Technology

Final Report, January 10, 1996

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Enclosure 4