VOGTLE ELECTRIC GENERATING PLANT UNITS 1 AND 2
INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(BACK-END)

ENCLOSURE 3

Vogtle Units 1 and 2
Technical Evaluation Report
on the Individual Plant Examination
Back-End Analysis

H. A. Wagage
J. F. Meyer

Prepared for the U.S. Nuclear Regulatory Commis
Under Contract NRC-05-91-068-42
November 1995

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E. EXECUTIVE SUMMARY

SCIENTECH, Inc., performed a review of the back-end portion of the Georgia Power Company's Vogtle Electric Generating Plant (VEGP) Units 1 and 2 Individual Plant Examination submittal.

E.1 Plant Characterization

The VEGP Units 1 and 2 each contain a Westinghouse-designed, 4-loop pressurized water reactor (PWR) with a large, dry containment. The containment structure is a horizontally and vertically prestressed, post-tensioned concrete cylinder resting on top of a reinforced concrete slab and enclosed by a prestressed, post-tensioned, concrete dome. The interior of the containment is lined with a 1/4-inch carbon steel liner plate. The containment has a free volume of 2.76E6 ft^3 and a median failure pressure of 137 psig. However, a lower bound value (5% probability) of 112 psig was used for the calculations.

E.2 Licensee's IPE Process

To assess the overall plant response to core damage initiators, the members of the IPE team took an integrated approach: they combined a probabilistic assessment of plant response to postulated initiating events with the use of a physical model to examine the effects of post-core-damage behavior and phenomenological uncertainties. They focused on sequences identified as dominant contributors to plant risk as well as on other sequences identified as important. They included in their assessment plant models and physical processes that reflected overall plant behavior before and after core damage.

Plant response trees (PRTs) were developed to consider all systems and operator actions, including containment functional events, that respond to an initiating event in order to prevent or mitigate release of radioactive materials from the containment.

The team did not develop containment event trees (CETs) to evaluate containment performance. MAAP analysis and phenomenological evaluation summaries were used to address phenomenological uncertainties surrounding containment performance and source-term assessment. The following general assumptions were used for the MAAP calculations:

- During the source-term analysis no credit was taken for accident mitigation activities because the back-end analysis assumed a 48-hour mission time (twice the IPE mission time of 24 hours) and yet found no containment failures;
- Based on evaluation of the VEGP containment/cavity design, the possibility of core debris leaving the cavity was assumed to be negligible;
- No failed equipment was recovered during the back-end analysis unless specifically defined in the front-end event trees;
- Accident sequences were evaluated by assuming that no more than two containment cooling units were operating; and

- No credit was taken for fission product retention in the auxiliary building.

The corporate office of the VEGP Southern Nuclear Operating Company (SNC) Project was responsible for the VEGP IPE program overall. An engineer from the SNC Nuclear Engineering and Licensing Department coordinated all IPE activities with the plant, corporate office, architect engineers, independent review group (IRG), and SNC Technical Services Department (TSD). The TSD was responsible for the management and technical oversight of the IPE program. Fauske & Associates, Inc., (FAI) performed the back-end analysis portion of the VEGP IPE.

The IPE team members performed a 1-day walkdown of the containment to familiarize themselves with it. In response to question 2 of the staff's RAI, the submittal notes that the models developed in the IPE represent the as-built, as-operated, as-maintained VEGP configurations as of January 1, 1991, with some exceptions, most of which relate to the front-end analysis. [2]

E.3 Back-End Analysis

The front-end analysis showed that each VEGP unit has a calculated containment damage frequency (CDF) of 4.9E-5 per year. This value represents credit taken for several front-end plant improvements by means of procedure enhancements scheduled for implementation. No back-end plant improvements were identified. The total CDF without these enhancements was 8.2E-5 per reactor year. About 70% of the total CDF resulted from loss of offsite power/station blackout (SBO) events. A majority of these events led to core damage as the result of diesel generator and decay heat removal system failures.

The IPE team defined "early" containment failure of the PWR as occurring before or within 6 hours of vessel breach, while NUREG-1150 defines it as occurring before vessel breach or within a few minutes afterward. This discrepancy became a moot point, however, since the VEGP back-end results indicated that no containment failures occurred during the 48-hour mission time.

The frequency of containment failure within 48 hours of mission time was zero, according to the results of the VEGP back-end analysis. The IPEs conducted for two other plants, Kewaunee and Point Beach, showed similar results. The IPEs for all three plants concluded that the early containment failures would not threaten the integrity of the containments if they were the result of any of the following: hydrogen combustion, direct containment heating, steam explosions, vessel blowdown, and thermal loading on penetrations. Molten core-concrete interactions and containment overpressurization are the only mechanisms identified that would cause containment failure after 48 hours. How the VEGP IPE team addressed phenomenological issues associated with severe accident progression is expected to be found in the phenomenological evaluation papers, which the submittal summarized but could not present in detail. A review of the papers themselves is outside the scope of this technical evaluation report.

Each of the VEGP units has a calculated uncontrolled fission product release frequency of 1.8E-6 per reactor year. This release frequency resulted from containment bypass caused by

steam generator tube rupture (SGTR) events. The conditional probability of containment bypass at VEGP was 3.4% of the total CDF, which is in the mid range compared with that of similar plants. The conditional probability of containment isolation failure at VEGP was 0.4%, which is in the low range compared with the other plants.
The key insights pertaining to the VEGP containment as noted by the licensee are as follows:

- The VEGP containment is able to remain intact for several tens of hours following core damage. This robustness allows natural deposition mechanisms to remove airborne fission products from the containment atmosphere, and provides adequate time for additional accident mitigation activities to be implemented;

- The primary system provides good fission product retention even after vessel failure during containment bypass sequences because of deposition on primary system structures;

- The VEGP cavity was expected to remain dry during most of the accident sequences. Water on the containment floor would have to reach a depth of 9 feet, 2 inches before it would spill into the reactor cavity. This would not be possible, even if all of the water in the refueling water storage tank were injected into the containment.

- The VEGP consists of two containment heat removal systems: eight containment cooling units (CCUs) and two containment spray system (CSS) trains. However, over the long term, only the CCU system is capable of removing decay heat from the containment. The containment spray system can only function as a short-term containment pressure reduction system.

E.4 Containment Performance Improvement Issues

One of the Containment Performance Improvement (CPI) Program recommendations that pertains to PWRs with large, dry containments is that utilities evaluate their containment and equipment vulnerabilities to hydrogen combustion (local and global) as part of their IPEs and that they identify the need for improvements in PWR procedures and equipment. Based on bounding analyses and conservative assumptions involving a worst-case SBO sequence, the IPE team concluded that hydrogen combustion could not threaten the VEGP containment integrity.

E.5 Vulnerabilities and Plant Improvements

Based on guidance provided in Generic Letter (GL) 88-20, Appendix 2, "Criteria for Selecting Important Severe Accident Sequences," the VEGP IPE team used the following screening criterion for determining sequences necessary to report to the NRC:

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

None of the VEGP functional core damage accident sequences exceeded the above source-term screening criterion. The IPE team subdivided SGTR sequences into four functional ones based on conditions that were not affecting accident progression. This obscured SGTR as a functional sequence that would lead to more than a 10% release of volatiles with a frequency greater than 1E-6 per reactor year. To determine whether a vulnerability existed that was related to unusually poor containment performance, the team used the following criterion:

- Any source-term analysis bin which represents containment failure, bypass, or failure to isolate, occurs with a frequency greater than 1E- events per year, and in which a single function, system, operator action, or other element can be identified which substantially contributes to the total frequency. The present state-of-the-art of containment systems analysis (as noted in Generic Letter 88-20) may be considered in evaluating any potential vulnerability identified by this criterion.

Based on the source-term release results, neither the containment bypass nor the failure to isolate frequency exceeded the limits of the above criterion. As already noted, the frequency of containment failure within 48 hours of mission time was zero. In summary, the VEGP back-end analysis identified no plant-specific vulnerabilities.

During the course of the VEGP IPE, several front-end plant improvements were identified as scheduled for implementation to reduce potential vulnerabilities. No back-end improvements were identified.

E.6 Observations

Overall, the VEGP IPE appears to have achieved the objectives of GL 88-20.

We take exception, however, to the team's subdivision of the SGTR sequences into four functional sequences based on conditions that were not affecting accident progression. This action obscured SGTR as a functional sequence that would lead to more than a 10% release of volatiles with a frequency greater than 1E-6 per reactor year. As a result, the IPE team identified "no sequences which resulted in a release of greater than that of PWR-4 with a frequency 1E-06 . . ." (from page 125 of the response to the staff's RAI). [2]

1. INTRODUCTION

1.1 Review Process

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

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

SCIENTECH sent the NRC a draft TER on the back-end portion of the VEGP IPE submittal in April 1995. Based in part on this draft submittal, the NRC staff submitted a Request for Additional Information (RAI) to Georgia Power Company (GPC) on June 16, 1995. GPC responded to the RAI in a document dated September 13, 1995. [2] This final TER is based on the original submittal and the response to the RAI.

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

1.2 Plant Characterization

The VEGP Units 1 and 2 each contain a Westinghouse-designed, 4-loop pressurized water reactor (PWR) with a large, dry containment. The containment structure is a horizontally and vertically prestressed, post-tensioned, concrete cylinder resting on top of a reinforced concrete slab and enclosed by a prestressed, post-tensioned, concrete dome. The containment has a free volume of 2.76E6 ft^3 and its interior is lined with 1/4-inch carbon steel liner plate. The principal dimensions of the containment building are listed in Table 4.1-1, page 4-2 of the submittal, and reproduced as Table 1 in this report.

Table 1. Summary of VEGP Containment Data

Type	Westinghouse Large, Dry
Architect/Engineer	Bechtel
Design Pressure	52 psig
Design Temperature	381¡F
Internal Free Volume	2.76E6 ft^3
Interior Diameter	140 ft
Interior Height	228 ft 9 in
Height to Spring Line	158 ft 9 in
Cylinder Wall Thickness	3 ft 9 in
Dome Thickness	3 ft 9 in
Liner Plate Thickness	1/4 in
Base Slab Thickness	10 ft 6 in
Cavity Floor Thickness	9 ft
Cavity Floor Area	648 ft^2
Height from Cavity Floor to Vessel Bottom	15 ft 6 in

The open design and significant venting areas for the subcompartments within the VEGP containment ensure a well-mixed atmosphere, which inhibits combustible gas pocketing. The lower compartment (between the containment floor and the operating deck) and the upper compartment (the large containment volume above the operating deck) are connected by large openings around the steam generators and above the reactor coolant pumps (RCPs). The lower compartment and the annular compartment (below the operating deck but outside the secondary shield wall) are connected by four manways on the 171-foot, 9-inch elevation. The annular and upper compartments are connected by steel gratings around the periphery of the operating deck.

The lower compartment contains four primary system loops inside the secondary shield wall. The annular compartment contains the four accumulator tanks, the pressurizer, and the pressurizer relief tanks outside the secondary shield wall.

The reactor cavity has a floor area of 648 ft^2, and remained dry during most of the accident sequences. Although water can easily drain from the upper compartment to the annular and lower compartments, the water would have to reach a depth of 9 feet, 2 inches, before it could spill into the reactor cavity through the cavity sump pump discharge pipe penetration. This would be unlikely to happen, even if the entire refueling water storage tank (RWST) volume (648,000 gallons of usable volume) were injected into the containment. The 10-foot, 6-inch basemat (9 feet at the cavity floor) is thicker than that of many other plants, which makes the molten corium-concrete interactions (MCCI) caused by basemat penetration relatively unlikely to occur at VEGP, also.

The VEGP consists of two containment heat removal systems: eight containment cooling units (CCUs) and two containment spray system (CSS) trains. The CCUs are capable of removing decay heat from the containment over the long term and, although it is not a containment system, the residual heat removal system also could be used to remove decay heat over the long term. The CSS, however, functions only as a short-term containment pressure reduction system.

The CCUs take suction from the upper compartment and discharge to the bottom of the containment building through four concrete ducts. Each duct consists of three large outlets located above the maximum containment flood elevation. The CCUs supply cool air to the regions around the reactor coolant pumps and steam generators, which flows upward through large openings in the operating deck floor.

Initially, the CSS takes suction from the RWST and injects into the containment through spray headers located under the containment dome. The CSS and emergency core cooling system pumps change to recirculation mode at the RWST low-low level alarm. In recirculation mode the CSS pumps take suction from the containment spray sumps and discharge to the spray headers. Potentially, the CSS reduces containment pressure in the short term, removes fission product from the containment atmosphere, and provides water into the containment by injecting from the RWST. Because the CSS is not equipped with heat exchangers, however, it alone cannot remove decay heat from the containment.

In recirculation mode the RHR pumps take suction from the two containment recirculation sumps, which are different from the containment spray sumps. The RHR heat exchangers cool the containment water before returning to the reactor coolant system (page 4-7 of the submittal).

2. TECHNICAL REVIEW

2.1 Licensee's IPE Process

2.1.1 Completeness and Methodology.

The VEGP IPE submittal contains a substantial amount of information with regard to the recommendations of Generic Letter (GL) 88-20, its supplements, and NUREG-1335. The submittal appears to be complete in accordance with the level of detail outlined in NUREG-1335. The methodology used to perform the IPE is described clearly, as are the team's basic underlying assumptions and the approach taken, which is consistent with the basic tenets of GL 88-20, Appendix 1. The important plant information and data are well documented and the key IPE results and findings are well presented.

The front-end analysis evaluated accident sequences over a 24-hour mission time. The IPE team defined core damage as occurring when the core temperature exceeds 1200°F (page 3-14 of the submittal).

To assess the overall plant response to core damage initiators, the members of the IPE team took an integrated approach: they combined a probabilistic assessment of plant response to postulated initiating events with the use of a physical model to examine the effects of post-core-damage behavior and phenomenological uncertainties. They focused on sequences identified as dominant contributors to plant risk as well as on other sequences identified as important. They included in their assessment plant models and physical processes that reflected overall plant behavior before and after core damage.

The IPE team developed plant response trees (PRTs) to consider all systems and operator actions, including containment functional events, which respond to an initiating event in order to prevent or mitigate release of radioactive materials from the containment. MAAP analysis provided the following information as input to the PRTs (Section 4.2, page 4-8):

- PRT top event success criteria;
- Timing of key events for human reliability analyses and for understanding sequence progression; and
- Accident sequence outcomes.

The IPE team did not develop containment event trees to evaluate containment performance. MAAP analysis and phenomenological evaluation summaries (PESs) were used to address phenomenological uncertainties about containment performance and to assess the source term. The MAAP calculations were made in the context of the following: (Section 4.2, page 4-9)

- During the source-term analysis no credit was taken for accident mitigation activities because the back-end analysis assumed a 48-hour mission time (twice the IPE mission time of 24 hours) and yet no containment failures were found;

- Based on an evaluation of the VEGP containment/cavity design, the possibility of core debris leaving the cavity was assumed to be negligible;

- No failed equipment was recovered during the back-end analysis except for that specifically defined in the front-end event trees;

- Accident sequences were evaluated subject to the assumption that no more than two CCUs were operating; and

- No credit was taken for fission product retention in the auxiliary building.

According to the submittal, the team supported PESs with available experimental information from the open literature and from the experiments performed at Fauske & Associates, Inc. (FAI). The FAI experiments provided information on direct containment heating (DCH) in a Zion-like geometry. The team performed Electric Power Research Institute (EPRI) MAAP sensitivity studies to address uncertainties not considered in the phenomenological evaluation summaries.

The IPE team defined "early" containment failure of the PWR as occurring before or within 6 hours of vessel breach, while NUREG-1150 defines it as occurring before vessel breach or within a few minutes afterward (page 2-13). This discrepancy became a moot point, however, since the VEGP back-end results indicated that no containment failures occurred during the 48-hour mission time.

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

Because VEGP is a dual-unit site, the IPE team examined the design of the two units and their dependencies. The major objectives of this examination were as follows:

- Determine whether the designs of the two units were substantially different to warrant performance of different IPEs;

- Identify and analyze inter-unit dependencies; and

- Identify and analyze any initiating event that could affect both units at the same time.

The methodology used to identify multi-unit effects at VEGP consisted of performing five types of analyses including plant familiarization and containment analyses that had an impact on the back-end analyses. Plant familiarization involved collecting and evaluating plant documentation on the design and operation of each unit; identifying unit dependencies among front-line systems, support systems, and the units; and performing plant walkdowns. Containment analyses for dual-unit considerations involved identifying differences and commonalities between the containment layout, systems, physical strength, and potential fission product release pathways of the two units. Most of the back-end information sources, including the containment structural analysis, cavity flooding analysis, and MAAP analyses were found to apply to both units. The team identified no differences between the units that would affect the containment or the source-term analyses.

The models developed in the IPE represent the as-built, as-operated, as-maintained VEGP configurations as of January 1, 1991, with some exceptions basically related to the front-end analysis. [2]

2.1.3 Licensee Participation and Peer Review.

The organizational structure of the VEGP IPE is presented in Figure 5.1-1, page 5-2 of the submittal, and is reproduced as Figure 1 of this report. The corporate office of the VEGP Southern Nuclear Operating Company (SNC) Project was responsible for the VEGP IPE program overall. An engineer from the SNC Nuclear Engineering and Licensing Department was designated to coordinate the IPE activities with the plant, corporate office, architect engineer support, independent review group (IRG), and the SNC Technical Services Department (TSD). The TSD was responsible for the management and technical oversight of the IPE program. FAI performed the back-end analyses of VEGP IPE.

Within the TSD PRA group, a senior-level engineer experienced in back-end analyses was responsible for performing technical reviews, participating in the containment walkdown, ensuring proper front-end back-end interfaces, and obtaining back-end technology transfer from FAI. The other TSD engineers were involved in performing the IPE on an as-needed basis. The PRA group supervisor was responsible for overseeing the IPE program including technical oversight of initial work scope document development, technical reviews, telephone conversations, and meetings.

A senior-level FAI engineer with PRA experience was dedicated to the VEGP back-end analyses in which several FAI engineers participated. FAI mandated and committed to the quality assurance requirements of 10 CFR Part 50, Appendix B.

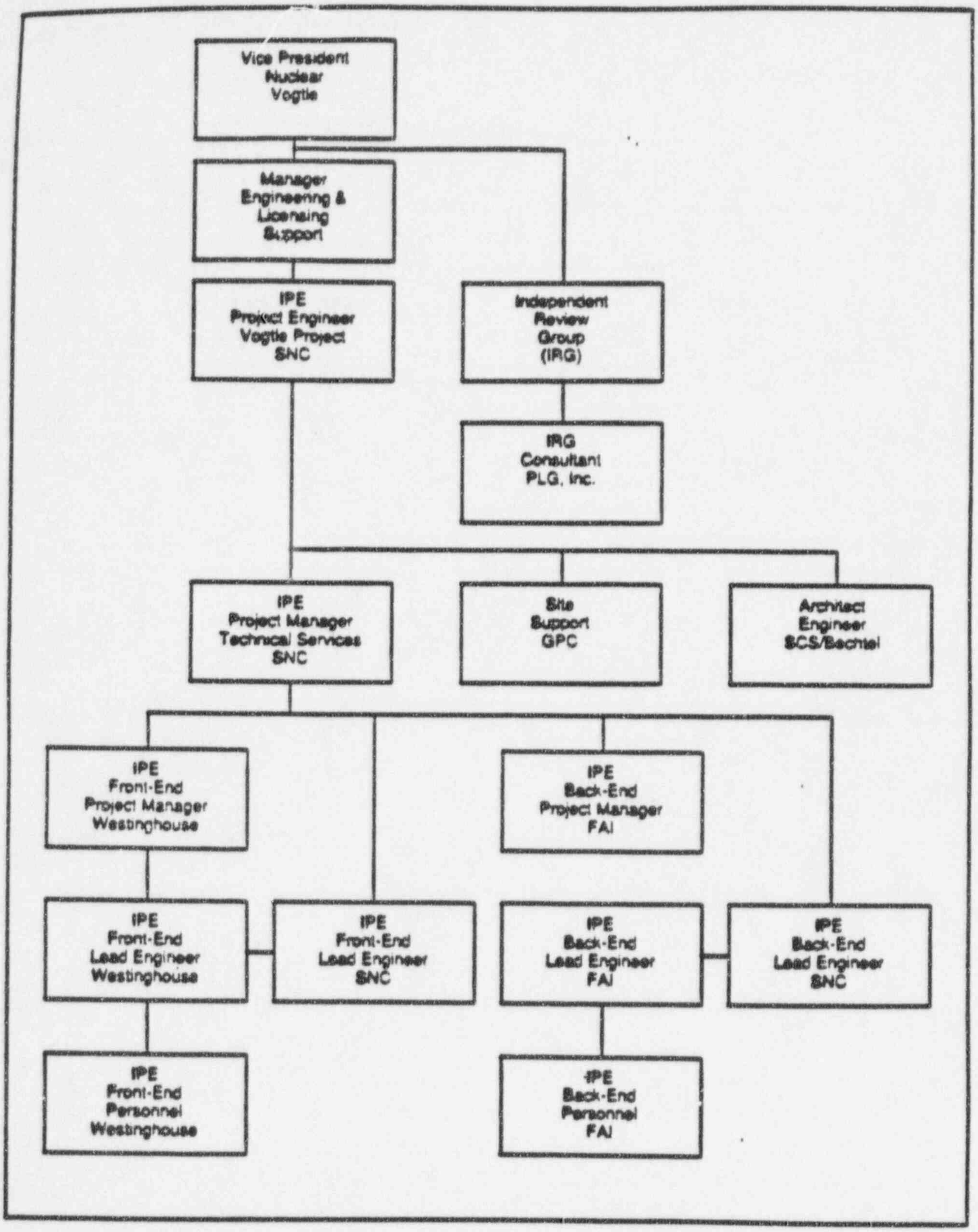

Figure 1. VEGP IPE organization functional structure

(reproduced from figure 5.1-1, page 5-2 of the submittal).

To help ensure that the results and conclusions of the IPE were applicable to and representative of VEPG, the SNC established an independent review group of personnel with expertise in diverse areas of the plant to review the IPE plan, process, results, and documentation. The IRG consisted of the following personnel:

- Plant staff: assistant manager, plant support (vice chairman of IRG); manager, training; manager, engineering technical support; manager, maintenance (acting);

- Corporate staff: manager, maintenance and support (Chairman of IRG); manager, engineering and licensing; manager, safety audit and engineering review;

- Southern Nuclear Operating Company: vice president, technical services; and

- PLG, Inc.: independent consultant.

The independent consultant from PLG, who was a senior-level staff member, assisted in the technical reviews and reviewed the application of the PRA methodology.

The IRG held several meetings in which the various organizational participants that prepared portions of the IPE presented related materials for review. IRG recommendations and/or outstanding questions that arose during these meetings became action items that were put on the agenda of subsequent meetings and were tracked until their resolution.

In a departure from the traditional process in which reviews are performed after the IPE is completely finished, site and corporate personnel reviewed the VEGP IPE incrementally at several intermediate stages in the analysis. In this way site and corporate personnel were introduced to the IPE process and many comments could be incorporated without affecting the orderly conduct of the analysis.

To this end, eight IRG meetings were held over a period of 15 months. Major agenda items relating to the back-end analysis were as follows:

- Containment pressure/heating;
- Hydrogen deflagration/detonation;
- Core-concrete interactions;
- Recovery process;
- Back-end analysis phenomenological papers;
- Front-end and back-end analysis results; and
- IPE NRC submittal report.

In response to the IRG comments, the IPE team revised the phenomenological evaluation summary on DCH to "indicate the areas considered in the evaluation of potential pathways for debris entrainment and to provide justification for including or excluding each potential area in the scope of analysis" (Table 5.3.1, page 5-15). The IPE team also included a comparison of the significant design aspect differences between the Zion and VEGP reactor cavities, concluding that DCH was less likely to occur at VEGP than at Zion. The major reasons were that at VEGP any core debris must make a 90-degree turn before exiting the instrument tunnel and the exit opening is well above the seal table.

By contrast, the Zion cavity/instrument tunnel opens directly into the lower compartment.

It appears that the licensee adequately participated in and reviewed the VEGP IPE.

2.2 Containment Analysis

2.2.1 Front-end Back-end Dependencies.

The IPE team postulated a total core damage frequency of 4.9E-5 per reactor year for VEGP. The dominant initiator for core damage was loss of offsite power (69.8%) followed by loss of coolant accidents (19.0%). The other contributors were transients (6.8%), steam generator tube ruptures (3.6%), special initiators (0.5%), and interfacing system LOCAs (0.1%). The special initiators consisted of the following: loss of instrument air; loss of nuclear service cooling water; loss of two 120 V AC vital instrument panels; loss of 125 V DC bus 1AD1 or 1BD1; loss of auxiliary component cooling water; and loss of one train of CB ESF electrical equipment room heating, ventilation, and air conditioning. Initiating-event contributors to core damage are shown in Figure 1.4-1, page 1-9 of the submittal, which is reproduced as Figure 2 in this report.

To assess containment performance and fission product release in the back-end analysis, the IPE team used sequence grouping to consolidate into a small number of plant damage states (PDSs) the large number of accident sequences that could lead to core damage.

Seven alphanumeric characters were used to identify characteristics important to understanding the accident sequences. The first two characters identified each accident initiator, as defined in Table 3.1-3, page 3-29 of the submittal, which is reproduced as Table 2 of this report.

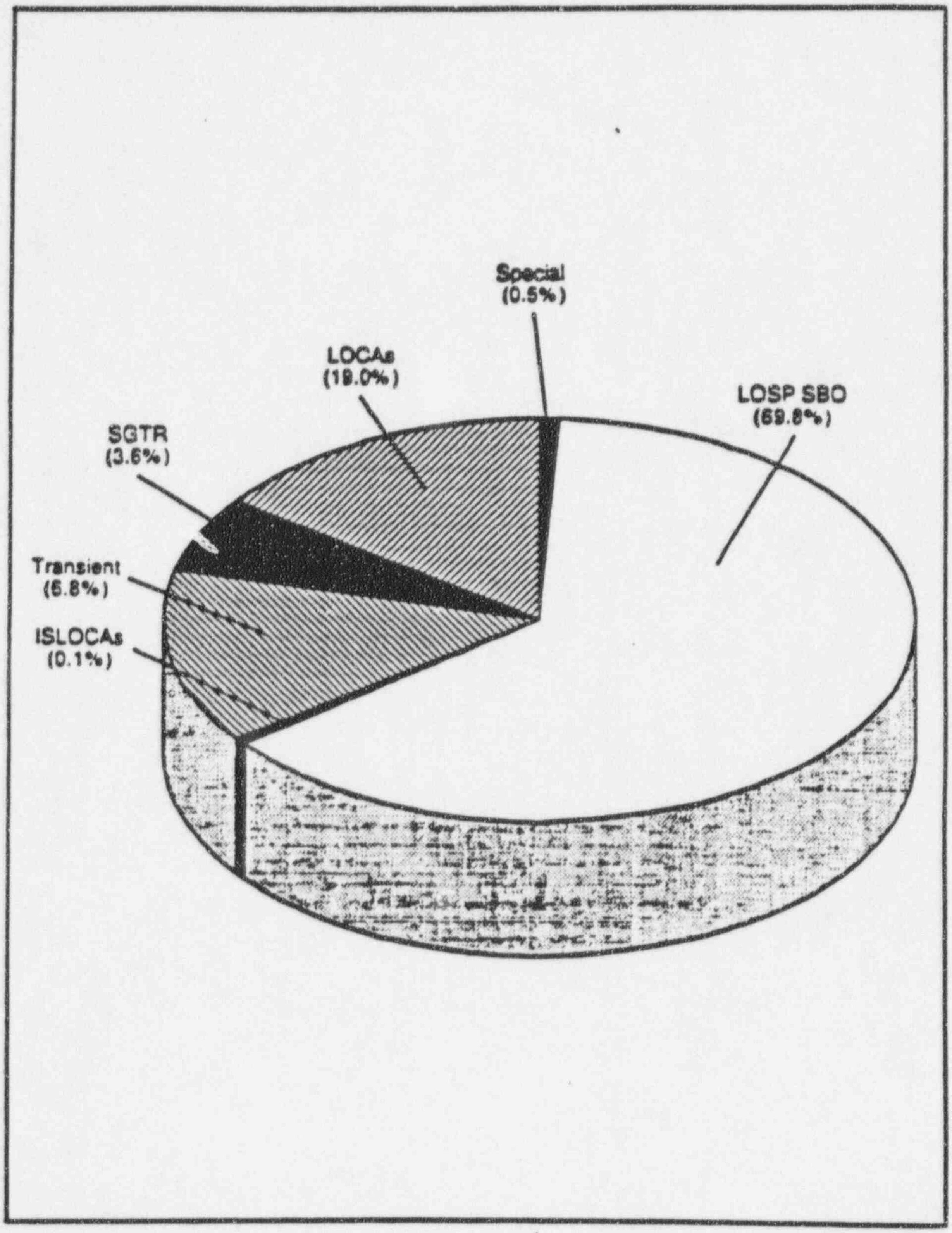

Figure 2. Core-damage frequency by initiating-event type

(reproduced from figure 1.4-1, page 1-9 of the submittal)

Table 2. Initiating Event Designators for Sequence Grouping

Designator	Initiating Event
LL	Large LOCA
ML	Medium LOCA
SL	Small LOCA
SG	SGTR
IL	Interfacing system LOCA
RV	Reactor vessel rupture
TR	Transients
BO	Station blackout
AT	Anticipated transient without trip

The third character identified the possible time intervals in which core damage might occur:

- Early core damage (occurs within 2 hours of accident initiation);
- InterMediate core damage (occurs within 2 to 6 hours of accident initiation); and
- Late core damage (occurs within 6 to 24 hours of accident initiation).

The fourth and fifth characters identified a combination of event tree top event conditions that identified the availability and modes of coolant injection/recirculation and CCU availability. These identifiers are listed in Table 3.1-4, page 3-30 of the submittal, which is reproduced in this report as Table 3.

The sixth character identified the state of the containment as one of the following:

- Containment is Isolated;
- Containment is Not isolated; or
- Direct containment Bypass.

The seventh character identified the RCS pressure when core damage occurred as either:

- High pressure; or
- Low pressure.

Table 3. Functional Failure Identifiers

Identifier	ECCS[1] Inject	ECCS Recirc	CCUs	Recirc HX	CS Inject	CS Recirc
00	Y	Y	Y	Y/N	Y	Y
01	Y	Y	Y	Y/N	Y	N
02	Y	Y	Y	Y/N	N	—
03	Y	Y	N	Y	Y	Y
04	Y	Y	N	Y	Y	N
05	Y	Y	N	Y	N	—
06	Y	Y	N	N	Y	Y
07	Y	Y	N	N	Y	N
08	Y	Y	N	N	N	—
09	Y	N	Y	—	Y	Y
10	Y	N	Y	—	Y	N
11	Y	N	Y	—	N	—
12	Y	N	N	—	Y	Y
13	Y	N	N	—	Y	N
14	Y	N	N	—	N	—
15	N	—	Y	—	Y	Y
16	N	—	Y	—	Y	N
17	N	—	Y	—	N	—
18	N	—	N	—	Y	Y
19	N	—	N	—	Y	N
20	N	—	N	—	N	—

(1) Acronyms:

ECCS INJECT – ECCS Injection from RWST
ECCS RECIRC – ECCS Recirculation
CCUs – Containment Cooling Units (at least 2/8)
RECIRC HX – Heat Removal via the RHR Heat Exchanger
CS INJECT – Containment Spray Injection from RWST
CS RECIRC – Containment Spray Recirculation

Of the PDSs defined using the above designators, 14 contributed more than 1% to the total CDF. The IPE team used the 14 PDSs in the back-end analysis. They are listed in table 3.4.2, pages 3-191 and 3-192 of the submittal, which is reproduced as Table 4 of this report.

It appears that the IPE team adequately considered front-end back-end dependencies through sequence grouping.

2.2.2 Containment Event Tree Development.

The IPE team concluded that separate containment event trees (CETs) were not applicable to the VEGP IPE and used plant response trees that incorporated CET aspects. PRTs are event trees that include the analysis of containment systems, which is traditionally performed in the back-end analysis. The output of a PRT includes the expected timing of core damage, the status of the emergency core cooling and containment heat removal systems, the state of the containment before core damage (e.g., bypassed, isolated, or not isolated) and the RCS pressure when core damage occurs. Various core damage states generally defined the end points of the PRT, which were used to quantify the source term.

The team developed a PRT for each initiating event identified and listed each tree in Table 3.1-1, page 3-6 of the submittal, which is reproduced in this report as Table 5.

2.2.3 Containment Failure Modes and Timing.

Bechtel Power Corporation performed a plant-specific structural analysis of the VEGP containment to determine its ultimate internal pressure capacity and its most likely failure locations. This analysis included the containment penetrations, hatches, encapsulation vessels, and the containment structure itself. The dominant failure modes were bending shear failure in the reinforced concrete containment basemat adjacent to the reinforced concrete cylinder wall at 145 psig, and failure of the welds at the emergency airlock bulkheads at 147 psig. The shell (hoop direction) wall and dome had capacities of 164 and 165 psig, respectively. The calculated containment fragility curve is illustrated in Figure 4.4-1, page 4-17 of the submittal, which is reproduced as Figure 3 of this report. The lower bound containment failure pressure (5% probability value) of 112 psig (127 psia) was used in determining containment overpressure failure. For VEGP, the expected overpressure failure area was 0.01 ft^2, based on trial-and-error MAAP results for a typical VEGP station blackout sequence.

Table 4. Core Damage Frequency by PDS

Damage State	Description	Frequency	Percent
BOE20IH	Station Blackout, early (0-2 hours) high-pressure core damage due to failure of all systems, containment isolated	1.75E-05	35.96
BOE01IH	Station Blackout, early (0-2 hours) high-pressure core damage with systems restored after core damage, containment isolated	9.39E-06	19.30
TRM11IH	Transient, intermediate (2-6 hours) high-pressure core damage without ECCS recirculation or containment spray injection, containment isolated	3.26E-06	6.69
TRE15IH	Transient, early (0-2 hours) high-pressure core damage without ECCS injection, containment isolated	3.18E-06	6.53
MLM11IL	Medium LOCA, intermediate (2-6 hours) low-pressure core damage without ECCS recirculation or containment spray injection, containment isolated	2.81E-06	5.76
BOM09IH	Station Blackout, intermediate (2-6 hours) high-pressure core damage without ECCS recirculation, containment isolated	2.18E-06	4.48
SLM18IL	Small LOCA, intermediate (2-6 hours) low-pressure core damage without ECCS injection or containment cooling units, containment isolated	1.46E-06	2.99
SLE17IH	Small LOCA, early (0-2 hours) high-pressure core damage without ECCS injection or containment spray injection, containment isolated	7.30E-07	1.50
SLL11IL	Small LOCA, late (6-24 hours) low-pressure core damage without ECCS recirculation or containment spray injection, containment isolated	7.28E-07	1.49

Table 4. Core Damage Frequency by PDS (cont'd.)

Damage State	Description	Frequency	Percent
SGE02BH	Steam Generator Tube Rupture, early (0-2 hours) high-pressure core damage without containment spray injection, containment is bypassed	6.79E-07	1.39
MLE18IL	Medium LOCA, early (0-2 hours) low-pressure core damage without ECCS injection or containment cooling units, containment isolated	6.67E-07	1.37
SGL11BH	Steam Generator Tube Rupture, late (6-24 hours) core damage without ECCS recirculation or containment spray injection, containment is bypassed	6.45E-07	1.32
MLL11IL	Medium LOCA, late (6-24 hours) low-pressure core damage without ECCS recirculation or containment spray injection, containment isolated	5.76E-07	1.18
LLM09IL	Large LOCA, intermediate (2-6 hours) low-pressure core damage without ECCS recirculation, containment isolated	5.35E-07	1.10

Table 5. Initiating Events Identified for VEGP and Their Frequencies
(reproduced from table 3.1-1, page 3-6 of the submittal)

Frequency (/Y)	Initiating Event
3.0E-04	Large Loss-of-Coolant Accident (LOCA)
8.0E-04	Medium Loss-of-Coolant Accident (LOCA)
6.6E-03	Small Loss-of-Coolant Accident (LOCA)
2.5E-02	Steam Generator Tube Rupture
5.4E-06	Interfacing Systems LOCA
1.0E-07	Reactor Vessel Rupture
8.0E-02	Positive Reactivity Insertion
1.2E-01	Loss of Reactor Coolant Flow
5.3E-01	Loss of Main Feedwater Flow
1.5E+00	Partial Loss of Main Feedwater Flow
3.5E-01	Loss of Condenser
7.3E-01	Turbine Trip
3.8E-02	Primary System Transient
6.9E-01	Reactor Trip
1.0E-02 4.1E-02	Loss of Offsite Power (Dual unit) Loss of Offsite Power (Single unit)
1.7E-01	Safety Injection Signal
2.6E-03 2.6E-03	Secondary Side Breaks (inside containment) Secondary Side Breaks (outside containment)
3.0E-02	Inadvertent Opening of Steam Valve
(a)	Station Blackout
(a)	Anticipated Transient Without Trip
2.4E-02	Loss of Instrument Air
1.4E-04	Loss of Nuclear Service Cooling Water
1.9E-03	Loss of Two 120 V AC Vital Instrument Panels
1.8E-03	Loss of 125 V DC Bus 1AD1 or 1BD1
1.3E-03	Loss of Auxiliary Component Cooling Water
4.1E-03	Loss of One Train of CB ESF Electrical Equipment Room HVAC

(a) These events are treated as consequential failures in the accident sequence analysis, and thus, no independent initiating event frequencies are calculated.

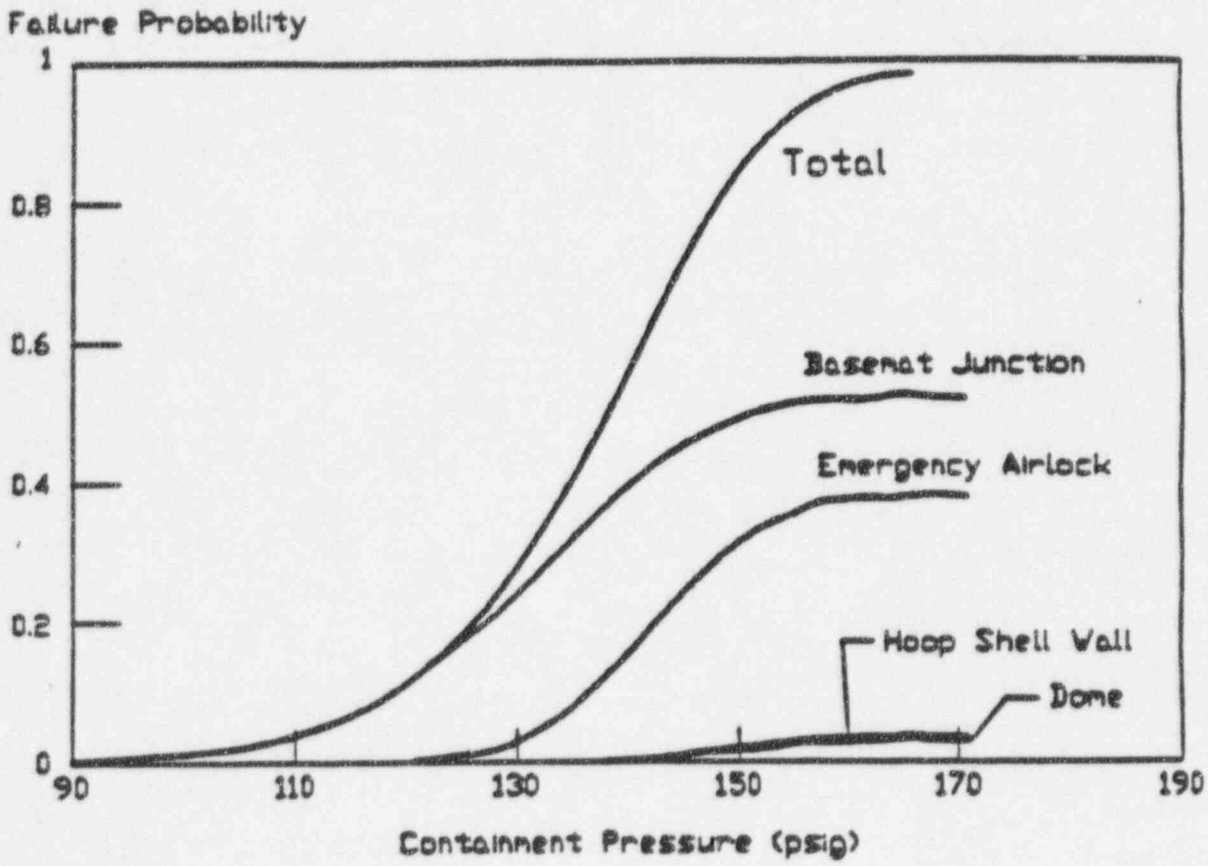

Figure 3. VEGP containment fragility curve
(reproduced from figure 4.4-1, page 4-17 of the submittal).

After performing phenomenological evaluations, the IPE team found that the following failure modes were unlikely to occur at VEGP:

Hydrogen Combustion. The team performed bounding analyses for a worst-case station blackout sequence using conservative assumptions. In assessing hydrogen deflagration they assumed in-core hydrogen generation due to 100% oxidation of all zirconium and metallic constituents of the lower core plate (about 2,200 pounds of hydrogen). The IPE team conservatively assumed that hydrogen would enter the containment at the time of core damage. Ignoring the presence of any passive or active heat sinks in the VEGP containment, the team assumed complete adiabatic isochoric combustion. Hydrogen was assumed to accumulate in the containment and burn at one time. The team calculated a post-burn containment pressure of 100 psia, which was below the lower-bound, 127-psia, ultimate capacity of the VEGP containment.

Hydrogen Detonation. The team concluded that hydrogen detonation by direct energy deposition was not possible in the VEGP containment because there were no potential ignition sources with sufficient energy to trigger such an event. The potential for deflagration-to-detonation transition (DDT) was evaluated using an engineering judgment procedure by Sherman and Berman. [3] Using this procedure the team assumed that the potential for DDT could be assessed based on a given mixture's intrinsic flammability (detonation cell width) and type of geometry. The VEGP analysis conservatively assumed a dry containment atmosphere and 100% oxidation of all zirconium and the lower core plate. This corresponded to a dry-basis hydrogen mole fraction of 14.3%. Based on the open design of the VEGP containment and its conduciveness to natural circulation, the containment gas was assumed to be uniformly mixed. The DDT assessment concluded that failure of the VEGP containment as the result of hydrogen detonation was very unlikely to occur.

Direct Containment Heating. The team performed a phenomenological evaluation to examine the likelihood of the VEGP containment to fail due to DCH. The team based its evaluation on DCH experiments and mechanistic models for debris dispersal which accounted for entrainment from the reactor cavity and de-entrainment at the instrument tunnel exit. The DCH methodology focused on the following:

- The debris mass that could be entrained in the reactor cavity and instrument tunnel; and
- The fraction of the entrained debris that could escape change in flow direction caused by the seal table enclosure.

Based on the results of the Argonne CWTI tests performed on a Zion-PWR-like geometry and based on the assumption that the VEGP cavity/instrument tunnel would trap more debris than Zion, the team posited that only 2% of the entrained debris would reach the annular compartment. [4] This percentage corresponded to a debris mass of 2,800 lb. In performing MAAP calculations the team assumed that 40% of the zirconium fuel cladding was oxidized before RPV failure, indicating that both DCH and hydrogen combustion would pressurize the VEGP containment to 75 psia. Without hydrogen burn, the containment pressure calculated was 50 psia.

Steam Explosions. Using an IDCOR methodology [5] the IPE team estimated that an in-vessel steam explosion was unlikely to occur with sufficient magnitude to fail the reactor vessel and consequently fail the VEGP containment. (Details of this analysis are not given in the submittal but are expected to be found in the phenomenological evaluation on steam explosion.)

The team calculated a containment pressure rise of 4 psig during an ex-vessel steam explosion. The calculated induced shock wave pressure at the containment wall was 56 psi. (We assume that such pressure on the cavity wall would be higher, but failure of the cavity wall at a magnitude sufficient to fail the containment was not described in the submittal.) Because these pressures were below the containment lower-bound capacity of 127 psig, ex-vessel steam explosions were considered unlikely to cause VEGP containment failure.

Molten Core-Concrete Interactions. The IPE team performed a bounding analysis to determine whether molten core-concrete interaction could lead to late containment failure. The team assumed that the concrete ablation rate was proportional to the total heat generation rate resulting from decay heat and chemical reactions. The sideward to downward ablation rate was assumed to be a constant equal to 0.2, which was lower than the best estimate value of 0.29 for long-term CCI experiments, and thus enhanced the downward ablation rate. Using MAAP results for a station blackout, which showed a cavity dryout time of 6 hours, the team calculated the basemat melt-through to occur at 53 hours after reactor trip. The team concluded the following (Section 4.4.2, page 4-20):

- Before MCCI-induced melt-through would occur, the containment would have failed due to other, relatively more rapid mechanisms;

- Before MCCI-induced melt-through would occur, mitigating actions would arrest containment failure; and

- Relative to other failure mechanisms, the source term for a basemat melt-through would be small because of the failure time (very late) and location (below ground).

Thermal Attack of Containment Penetrations. The IPE team reviewed the VEGP containment penetrations to assess the potential for their direct contact by core debris. The team analyzed data on the nonmetallic seal materials [6] used in the VEGP penetrations along with the results of environmental qualification work to determine the responses of the penetrations to expected worst-case severe accident conditions. The limiting sealant material at VEGP was found to be a polysulfene thermoplastic.

The team found that the mechanical and electrical penetrations would not come into contact with core debris that would be dispersed during a high pressure melt ejection (HPME) because most of the debris would be removed at the seal table enclosure and because no paths existed for such debris to come into direct contact with the containment penetrations. Therefore, the team concluded, thermal loading on penetration nonmetallic materials would not cause degradation and leakage from the containment in accidents postulated for VEGP. Also, the IPE team concluded that gas temperature in the containment would not increase above the operational limits of the seals (page 4-21 of the submittal).

Vessel Thrust Forces. The IPE team estimated that 1E6 lb was the maximum jet thrust force that could be expected during the expulsion of molten core debris through the failed vessel. This force was below the lower-bound dead weight of the RPV (excluding the combined weight of fuel cladding, control rods, and lower core plate), which was estimated to be 1.3E6 lbf. Therefore, the thrust force was found not sufficient to lift the vessel and internals.

The IPE team found that the following failure modes were likely to occur at VEGP:

Containment Overpressure. Overpressure might fail the containment in severe accident scenarios where containment heat removal was not available as a result of failure to inject the RWST, failure to align for recirculation, and failure of all CCUs.

The VEGP cavity was expected to remain dry during most of the accident sequences. Water on the containment floor would have to reach a depth of 9 feet, 2 inches, before it would spill into the reactor cavity. This was not considered possible even if all of the water in the RWST were injected into the containment. Also, nearly all core debris was expected to remain in the cavity. Therefore, the team concluded, containment pressurization would occur mainly as the result of noncondensible gas generation.

Containment Isolation Failure. See Section 2.2.4 of this report.

Containment Bypass. This could result from the failure of the pressure boundary between the high-pressure RCS and a lower pressure line that penetrated the containment. Containment bypass was postulated to be an accident initiator that could result in core damage. ISLOCA and SGTR were shown to be the most likely causes of bypass, both of which contributed 3.4% to the total VEGP core damage frequency.

It appears that the IPE team considered all of the containment failure modes listed in Table 2 of NUREG-1335. Details on how the VEGP IPE team addressed phenomenological issues associated with severe accident progression are expected to be found in the phenomenological evaluation papers.

2.2.4 Containment Isolation Failure.

The IPE team identified containment isolation failure as a possible mode of containment failure at VEGP. Such an isolation failure would be the result of a mechanical or operational failure to close containment fluid system penetrations, which would enter the containment before or after initiation of core damage, in order to limit fission product release to the auxiliary building or to the environment. Containment isolation was assumed to fail as the result of all check valves in fluid lines failing to close (Section 4.4.3, page 4-23) and as the result of one or more of the following conditions:

- A fluid line or mechanical penetration that is required to be closed during power operation has been left unisolated;
- A fluid line, which has isolation valves that are required to close on an isolation signal, fails to close; and
- A fluid line, which is part of a safety system and is required to remain open following the generation of isolation signals, is not closed by the operators if the system is "failed" or the operation of the system is terminated.

Critical containment penetrations that could cause isolation failures and result in significant fission product releases to the environment were identified based on meeting either of the following screening criteria (Section 4.4.3, page 4-23):

- The line penetrating the containment is a containment sump or reactor cavity sump drain line; and/or
- The line penetrating the containment is more than 2 inches in diameter, is connected directly to the containment atmosphere, and is not part of a closed system outside the containment capable of withstanding severe accident conditions.

Failure of containment isolation was modeled as a PRT top event. The IPE team found that no accident sequences would involve isolation failures with significant fission product release. The frequency of isolation failures at VEGP was calculated to be 1.68E-7 per reactor year, which constituted 0.4% of the total core damage frequency.

2.2.5 System/Human Response.

The following back-end operator actions were modeled in the VEGP PRT:

OAS OA to establish containment spray recirculation; and

OCI OA to isolate containment.

The primary goal of the OA OAS was to maintain containment integrity to prevent/mitigate escape of fission products to the environment. The intermediate supporting goals were to avoid containment overpressurization from hydrogen combustion or steam concentration. Other intermediate goals were to provide fission product scrubbing within the containment, and to cool the containment atmosphere and control sump pH. The immediate objective was to realign the spray suction from the RWST to the containment sump in time to prevent cavitation of the spray pumps. In order for the OA to succeed, at least one containment spray pump would have to be running and aligned to take suction from the containment sump, as noted on page 86 of the response to the staff's RAI. [2]

The primary goal of the OA OCI was to limit any offsite dose, should core damage occur as a result of the event. The immediate objective of this action was to verify containment Phase A isolation, and if isolation had *not* occurred, to manually actuate isolation. In order for the OA to succeed, the operator would need to identify the failure of containment isolation and manually actuate containment Phase A isolation, as noted on page 65 of the response to the staff's RAI. [2]

To model the above operator actions, the IPE team used the following human error probabilities calculated with SLIM methodology (see Table 3.3.3-2 of the submittal):

Establish containment spray recirculation	4.73E-4
Establish containment spray recirculation, contingent on failure of the OA to establish recirculation	1.50E-1
Isolate containment: No power - manual alignment	6.98E-3
Isolate containment: With power - auto alignment	1.00E-3

2.2.6 Radionuclide Release Categories and Characterization.

Based on the guidance provided in GL 88-20, Appendix 2, the IPE team used the following functional screening criteria to identify which severe accident sequences should be considered in the back-end analysis:

- Any functional sequence that contributes 1E-6 or more per reactor year to the core damage frequency;
- Any functional sequence that contributes 5% or more to the total CDF;
- Any functional sequence that has a core damage frequency greater than or equal to 1E-6 per reactor year and that leads to containment failure, which can result in a radioactive release magnitude greater than or equal to the PWR-4 release category of WASH-1400;
- Functional sequences that contribute to a containment bypass frequency in excess of 1E-7 per reactor year; and

- Any functional sequences judged to be important contributors to core damage frequency or poor containment performance.

Using the above criteria the IPE team selected 25 functional core damage sequences for the back-end analysis. These 25 sequences were binned into 14 source-term bins according to their expected source-term characteristics. The key functions that were important for radiological releases from the VEGP containment were the containment status, the availability of debris cooling and containment heat removal, and the RCS pressure at vessel failure. The team used 14 analyzed sequences to represent 25 sequences; 11 sequences were bounded by the analyzed sequences. Using MAAP calculations the team assigned source terms to the 14 source term bins.

The team defined radionuclide release categories based on containment failure timing (early or late), containment failure mode (overpressure, not isolated, or bypass), and airborne fractional release of fission products to the environment. After assigning appropriate release categories to the 14 analyzed sequences the team found that the VEGP source-term results fell into four categories (Table 4.7-4, pages 4-37 and 4-38):

- Release Category A (76.1% of total CDF): No containment failure within 48-hour mission time, but failure could eventually occur without further mitigating action; noble gases and less than 0.1% volatiles released;

- Release Category S (20.1% of total CDF): Success. Leakage only, successful maintenance of containment integrity; containment not bypassed; isolation successful before core damage;

- Release Category T (3.4% of total CDF): Containment bypassed with noble gases and more than 10% volatiles released; and

- Release Category G (0.4% of total CDF): Containment failure before vessel failure with all noble gases and up to 10% of the volatiles released (containment not isolated).

The VEGP back-end results showed that steam generator tube rupture would lead to more than a 10% release of volatiles with a frequency of 1.56E-6 per reactor year. However, the IPE defined four SGTR functional sequences (i.e., SGE02BH, SGL11BH, SGL20BH, and SGE15BH) which differ based mainly on their core and containment cooling status, as defined in Table 3.1-4, page 3-30 of the submittal. As explained on page 124 of the response to the RAI, the IPE team recognized that the containment cooling status was unimportant for SGTR sequences but, to be consistent with other PDSs, applied the de[illegible]itor reflecting the ECCS as well as containment heat removal for SGTR sequences. [2] [illegible]ne four PDSs of functional sequences (i.e., SGE02BH, SGL11BH, SGL20BH, and SGE15BH) were later combined as a single source-term bin.

We take exception to the team's subdivision of the SGTR sequences into four functional sequences based on conditions that were not affecting accident progression. This action obscured SGTR as a functional sequence that would lead to more than a 10% release of volatiles with a frequency greater than 1E-6 per reactor year. As a result, and as noted on page 125 of the response to the RAI, the IPE team identified "no sequences which resulted in a release of greater than that of PWR-4 with a frequency 1E-06 . . ." [2]

2.3 Quantitative Assessment of Accident Progression and Containment Behavior

2.3.1 Severe Accident Progression.

Section 4.6, pages 4-24 and 4-25 of the submittal, describes how the IPE team analyzed severe accident progression at VEGP. The team assumed no in-vessel recovery would occur at VEGP. In sequences with low pressure at vessel failure, all of the core debris was assumed to remain in the cavity. In sequences with high pressure at vessel failure, HPME was assumed to occur, resulting in an entrainment of 2% of the core debris, which then participated in DCH. The basis for this entrainment value is provided in FAI/91-122, "Vogtle Electric Generating Plant Units 1 and 2 Phenomenological Evaluation Summary on Direct Containment Heating in Support of the Individual Plant Examination" of which a brief summary is given in the submittal. [1] The IPE team used a model developed by Walker for Zion in estimating the likelihood of debris particles not deflecting with the flow as the result of a change in flow direction, thus affecting the structural boundaries of the flow path. [7]

The Walker model accounted for a single 90° turn that core debris would have to make to escape into the lower compartment. Before exiting the instrument tunnel to the annular compartment, the debris would have to make two 90° turns. The concrete platform that extends part of the way across the instrument tunnel and forms a "lip" would enhance debris de-entrainment further. Therefore, the IPE team considered the 2% value obtained using the Walker model to be a conservative one.

Section 2.2.3 of this report describes how the IPE team evaluated severe accident threats to the VEGP containment.

2.3.2 Dominant Contributors: Consistency with IPE Insights.

The results of the front-end analysis of the IPE showed that each VEGP unit had a calculated core damage frequency of 4.9E-5 per year. As shown in Tables 6 and 7, the CDF calculated for VEGP is in the mid range compared with that for other plants whose VEGP results SCIENTECH examined. About 70% of the total CDF at VEGP resulted from loss of offsite power/station blackout events. Most of these events led to core damage because of the failures of the diesel generators and decay heat removal systems.

The results of the back-end analysis of the IPE showed that the containment failure frequency within 48 hours of mission time was zero. As shown in Tables 6 and 7, the IPEs conducted of the Kewaunee and Point Beach plants produced similar results. With respect to all three plants each IPE concluded that early containment failures resulting from the following would not threaten the integrity of the containments: hydrogen combustion, direct containment heating, steam explosions, vessel blowdown, and thermal loading on penetrations. Molten core-concrete interactions and containment overpressurization were the only mechanisms identified that would cause containment failure and these would occur sometime after the 48-hour mission time.

Each of the VEGP units had a calculated uncontrolled fission product release frequency of 1.8E-6 per reactor year. This release frequency would result from containment bypass caused by steam generator tube rupture events. The conditional probability of containment bypass at VEGP was 3.4% of the total CDF, which is in the mid range compared with the probabilities at the other plants considered, except Zion, where it was 30% (Table 6). The conditional probability of containment isolation failure at VEGP was 0.4%, which is in the low range compared with the other plants.

2.3.3 Characterization of Containment Performance.

As noted in Section 2.2.2 of this report, the VEGP IPE team did not develop CETs but modeled containment performance into PRTs developed during the front-end analysis. The output of PRTs included the expected timing of core damage, the status of the emergency core cooling and containment heat removal systems, the state of the containment before core damage (such as whether it was bypassed, isolated, or not isolated) and the Reactor Coolant System pressure when core damage occurred (Section 3.1.4, page 3-16). Various damage states generally defined the end points of the PRTs, and were used subsequently for determining the source term.

Table 6. Conditional Containment Failure Probability during Mission Time (Percent)

Study	CDF per rx yr	Early Failure	Late Failure	Bypass	Isolation Failure	Intact
Diablo Canyon IPE	8.8E-5	4.6	45.2	1.8	7	41.4
Maine Yankee IPE[1]	7.4E-5	8	48	2.1	*	43
Palo Verde IPE	9.0E-5	10	14	4	0[2]	72
Kewaunee IPE	6.6E-5	0	0	8	0.023	92
Zion IPE	4.0E-6	0	5	30	2	63
Haddam Neck IPE	1.8E-4	0.18	54	6.5	0.5	39
Point Beach IPE	1.0E-4	0	0	6.1	0.031	94
Farley IPE	1.3E-4	0	3.1	0.36	0.06	96.4
Zion/NUREG-1150	6.2E-5	1.5	25	0.5	na	73
San Onofre IPE	3.0E-5	0	9.4	6.7	0.07	83.8[3]
Vogtle	4.9E-5	0	0	3.4	0.4	96.2

Table 7. Conditional Containment Failure Probability Beyond Mission Time (Percent)

Study	CDF rx yr	Early Failure	Late Failure	Bypass	Isolation Failure	Intact
Diablo Canyon IPE	8.8E-5	4.6	66.6	1.8	7	20
Maine Yankee IPE[1]	7.4E-5	8	48	2.1	*	43
Palo Verde IPE	9.0E-5	10	14	4	0[2]	72
Kewaunee IPE	6.6E-5	0	49	8	0.023	43
Zion IPE	4.0E-6	0	5	30	2	63
Haddam Neck IPE	1.8E-4	0.18	54	6.5	0.5	39
Point Beach IPE	1.0E-4	0	17.4	6.1	0.031	76.6
Farley IPE	1.3E-4	0	96.2	0.36	0.06	3.3
Zion/NUREG-1150	6.2E-5	1.5	25	0.5	na	73
San Onofre IPE	3.0E-5	0	9.4	6.7	0.07	83.8[3]
Vogtle	4.9E-5	0	76.1	3.4	0.4	20.1

Legend for Tables 6 and 7

* Bypass and isolation combined
na not available
[1] Values do not add up to "100."
[2] Probability is less than 0.001, conditional on core melt.
[3] Includes MCCI basemat penetration failures

PRTs were developed for the following initiating events: large LOCA, medium LOCA, small LOCA, steam generator tube rupture, secondary side breaks, transients, anticipated transient without trip, station blackout, and special initiators. Special initiating events included those resulting from failures of plant systems that may cause a reactor trip while adversely affecting the ability of plant systems to respond to such events. These PRTs are presented in Appendix A of the submittal.

PRTs modeled the responses of plant systems and operators to initiating events and the sequence of system and operator response. In general, each PRT reflected the responses of and the dependencies between system and operator actions (e.g., an operator action may only be needed if an emergency safety feature system fails to actuate). PRT sequences progressed to one of three end points: success, indicating that no core damage had occurred; a transfer to another PRT, resulting from a consequential failure of a plant system (e.g., failure of reactor trip would lead to the ATWT PRT) or core damage and the resultant damage states.

As described in Section 3.1.3, page 3-14 of the submittal, the PRT top event success criteria were based on a number of sources, including the VEGP Final Safety Analysis Report, various Westinghouse Owner's Group generic technical reports, and the Emergency Operating Procedures. Plant-specific calculations using the MAAP, TREAT, or COMPACT computer codes were performed when the required information was not available in the above sources.

2.3.4 Impact on Equipment Behavior.

As described in Section 4.1.2, page 4-6 of the submittal, the VEGP design includes the following two containment heat removal systems: eight containment cooling units and two containment spray units. However, only CCUs were designed to remove decay heat (long term) from the containment for design basis accidents. CSUs served as a short-term containment pressure reduction system. The IPE team concluded that only the CCUs needed to be assessed for containment equipment survivability.

Each VEGP CCU consisted primarily of a JOY AXIANE fan, which was directly coupled to a 150/75 horsepower Reliance electric motor. This motor had Class H, Type RN insulation, which was designed for 3 to 4 hours of emergency mode operation, at 350°F ambient temperature, with gradual reduction to 250°F, and ambient pressure of 85 psig (100 psia). Because the MAAP results showed that under analyzed accident sequences the containment pressure and temperature would not exceed the above limits, the team concluded that CCUs would function under analyzed accident conditions.

It appears that the IPE team has adequately considered equipment vulnerability to severe accident conditions expected at VEGP.

2.3.5 Uncertainty and Sensitivity Analysis.

The IPE team used a three-part approach to address phenomenological uncertainties associated with the back-end analysis. The first approach was to address the phenomena identified in NUREG-1335 by performing evaluations, summaries of which appear in the submittal. (See Section 2.2.3 of this report.) These phenomenological evaluations mainly concerned postulated early containment failure mechanisms, core-concrete interaction, and containment fragility. Phenomenological evaluations were performed to address the uncertainties of the following (Table 4.4-1, pages 4-12 and 4-13):

- Hydrogen combustion;
- Direct containment heating;
- Steam explosions;
- Molten core-concrete interactions;
- Vessel blowdown;
- Thermal loading on penetrations;
- Overpressurization;
- Containment isolation failure; and
- Containment bypass.

The second approach was to perform MAAP sensitivity studies recommended by the Electric Power Research Institute [8] to address the following phenomenological uncertainties that were not considered in the evaluations (pages 4-60 through 4-65 of the submittal). [1]

Hydrogen Production and Combustion in Containment. The effect was assessed of the MAAP input flame flux multiplier, "FLPHI," on hydrogen burn completeness. The base case calculation used a value of 2 for FLPHI. Increasing this value to 10 showed results identical with those of the base case.

The team also assessed the effect of the core blockage model on cladding oxidation and hence on in-vessel hydrogen production. MAAP's core blockage model does not allow oxidation and gas flow through core nodes after onset of melting in that node. Although the core blockage model was not used in the base case calculations, it was used in the sensitivity case where cladding oxidation was reduced to 33% compared with 45% for the base case. The source term was found not to be sensitive to the core blockage model.

Jet burning and auto-ignition of hydrogen had only a minor effect on the amount of hydrogen burned and no effect on the source term. The sensitivity case that the team ran without jet burning and with the auto-ignition turned off resulted in 226 pounds less of hydrogen burn than in the base case.

Hot Leg Creep Rupture Failure. In the base case sequence, an elevated hot leg/surge line temperature up to 1340°F occurred before vessel failure, while the RCS pressure was at 2350 psia. These RCS conditions were in the low-end range of potential creep rupture failure of the hot leg. The team ran a sensitivity case by assuming the occurrence of a hot leg failure with a 1 ft^2 break size before vessel failure when the hot leg temperature exceeded 1340°F. Failure of the hot leg converted a high-pressure sequence into a low-pressure sequence because of depressurization through the break. RCS depressurization enabled injection from the accumulators, which delayed slightly the vessel failure. Although the primary system retained more volatile fission products, the increase in source term from the containment was minimal.

Reduced Ex-Vessel Debris Coolability. Reduced ex-vessel debris coolability had no effect on accident progression timing and a negligible effect on the source term.

The third approach that the IPE team took was to perform MAAP sensitivity studies to address additional uncertainties that were identified during the VEGP source-term evaluation. Additional sensitivity cases were as follows:

Isolation Failure Area. The containment penetrations with the highest probability of failing during the most likely failure-to-isolate sequence (a station blackout) at VEGP were the 8-inch normal containment purge supply line, the 8-inch normal containment purge exhaust line, and the 3-inch normal containment sump pump discharge line. All of these lines had the same failure probability of 2.83E-3 per reactor year. The base case assumed an 8-inch purge line failure would occur and cause isolation failure, assuming the 3-inch line failure in the sensitivity case did not affect the accident progression significantly. However, 410 pounds more hydrogen burned and the peak containment pressure rose 14 psi higher than in the base case.

Bypass Flow Area. The team assumed for the base case SGTR sequence a tube area equivalent to 200% of the cross-sectional flow area of a single tube (4.03E-3 ft^2). To account for line losses not accounted for in the MAAP model, the IPE team assumed an area for the sensitivity case of half that size (i.e., 2.02E-3 ft^2). The smaller break area had a large impact on sequence timing and source-term release. The initiation of core melt was delayed 10 hours and RWST inventory was conserved. The source term was reduced with volatile release decreasing from 41 to 18%.

The team modeled the base case ISLOCA as an 0.1-ft^2 break in the hot leg outside containment, based on an upper bound of 0.1 ft^2 for the RHR pump seals (both pumps). Two sensitivity cases were run by considering a smaller flow area (0.034 ft^2 for a 2.5-inch break) and a larger flow area (0.545 ft^2 for a 10-inch break, which reflected the circumferential failure of the RHR heat exchanger exit piping). The larger ISLOCA sequence had a 2-hour accelerated sequence progression but the source-term results were similar to those of the base case. The smaller ISLOCA sequence had a delayed sequence progression and a reduced source term resulting from increased fission product retention in the primary system.

RWST Refill. For the base case source-term calculations, the IPE team did not take credit for recovery actions, such as adding more water to the containment. The base case was run assuming that ECCS injection continued until 144,000 gallons (9% usable volume) remained in the RWST from its initial inventory of 715,000 gallons. Spray circulation was successful. The team ran the sensitivity case assuming that an additional 300,000 gallons of water was added to the RWST at 10 hours, at which time spray circulation stopped and ECCS injection (into the failed vessel) resumed until the water ran out. The additional inventory flooded the cavity and overflowed into the containment. Concrete ablation and combustible gases burning in the cavity were terminated. The source-term release at 48 hours was nearly identical to that of the base case, but a safe, stable state had been achieved.

It appears that the IPE team adequately addressed the phenomenological uncertainties associated with accident progression. Some details are expected to be found in the phenomenological evaluation papers, a review of which is outside the scope of this work.

2.4 Reducing Probability of Core Damage or Fission Product Release

2.4.1 Definition of Vulnerability.

Based on guidance provided in Generic Letter 88-20, Appendix 2, "Criteria for Selecting Important Severe Accident Sequences," the VEGP IPE team used the following screening criterion for determining the important sequences that might lead to radioactive releases for reportability to the NRC (Section 4.9, page 4-69):

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

None of the VEGP functional core damage accident sequences exceeded the above source-term screening criterion.

To determine whether a vulnerability existed that was related to unusually poor containment performance, the team used the following criterion:

- Any source-term analysis bin which represents containment failure, bypass, or failure to isolate, occurs with a frequency greater than 1E-5 events per year, and in which a single function, system, operator action, or other element can be identified which substantially contributes to the total frequency. The present state-of-the-art of containment systems analysis (as noted in Generic Letter 88-20) may be considered in evaluating any potential vulnerability identified by this criterion.

Based on the source-term release results, neither the containment bypass nor the failure to isolate frequency exceeded the above criterion limit (Table 4.7-7 of the submittal). Also, the frequency of containment failure within the 48-hour mission time was zero. In summary, no plant-specific vulnerabilities were identified in the VEGP back-end analysis.

2.4.2 Plant Improvements.

During the course of the VEGP IPE, several front-end plant improvements were identified and had been scheduled to be implemented to reduce potential vulnerabilities. No back-end improvements were identified.

2.5 Responses to CPI Program Recommendations

One of the Containment Performance Improvement (CPI) Program recommendations that pertains to PWRs with large, dry containments is that utilities evaluate their containment and equipment vulnerabilities to hydrogen combustion (local and global) as part of their IPEs and that they identify the need for improvements in PWR procedures and equipment.

Section 4.4.2, pages 4-14 and 4-15 of the submittal, describes the phenomenological evaluations of hydrogen deflagration and detonation, which the IPE team performed for the VEGP. Based on bounding analyses and conservative assumptions involving a worst-case station blackout sequence, the IPE team concluded that hydrogen combustion could not threaten the VEGP containment integrity.

2.6 IPE Insights, Improvements and Commitments

The key insights pertaining to the VEGP containment as noted by the licensee are:

- The VEGP containment is able to remain intact for several tens of hours following core damage. This robustness allows natural deposition mechanisms to remove airborne fission products from the containment atmosphere, and provides adequate time for additional accident mitigation activities to be implemented;

- During containment bypass sequences the primary system provides good fission product retention even after vessel failure because of the deposition of core debris on primary system structures;

- The VEGP cavity was expected to remain dry in most of the accident sequences. Water on the containment floor would have to reach a depth of 9 feet, 2 inches, before it would spill into the reactor cavity. This would not be possible even if all of the water in the refueling water storage tank were injected into the containment; and

- The VEGP consists of two containment heat removal systems: eight CCUs and two containment spray system trains. However, only the CCU system is capable of removing decay heat from the containment over the long term. The containment spray system can function only as a short-term containment pressure reduction system.

The licensee identified no back-end improvements and made no commitments to undertake any.

3. OVERALL EVALUATION, CONCLUSION, AND INSIGHTS

By conducting a PRA, the Georgia Power Company sought to find vulnerabilities that might exist at VEGP in the event of severe accidents and internal flooding, and to recommend cost-effective safety improvements that would reduce or eliminate the most important risk contributors identified. The front-end analysis showed that each VEGP unit had a calculated core damage frequency of 4.9E-5 per year. This value represents credit taken for several front-end plant improvements by means of procedure enhancements, which had been scheduled for implementation. No back-end plant improvements were identified. Without these enhancements the total CDF for each VEGP unit was 8.2E-5 per reactor year. About 70% of the total CDF resulted from loss of offsite power/station blackout events. Most of these events would lead to core damage from the failure of the diesel generators and decay heat removal systems.

The results of the back-end analysis of the IPE showed that the containment failure frequency within 48 hours of mission time was zero. IPEs conducted at Kewaunee and Point Beach produced similar results and at all three plants it was concluded that the following would not threaten the integrity of the containments by causing early failures: hydrogen combustion, direct containment heating, steam explosions, vessel blowdown, and thermal loading on penetrations. Molten core-concrete interactions and containment overpressurization were the only conditions identified that would cause containment failure and neither of these would occur until sometime after the 48-hour mission time. Details of how the IPE addressed phenomenological issues of severe accident progression are expected to be found in the phenomenological evaluation papers, which the submittal summarizes, but a review of which is outside the scope of this work.

Each of the VEGP units had a calculated uncontrolled fission product release frequency of 1.8E-6 per reactor year. This release frequency resulted from containment bypass caused by steam generator tube rupture events. Of the total CDF at VEGP, the conditional probability of containment bypass was 3.4%; for containment isolation failure it was 0.4%. Compared with the contributors to the CDFs of similar plants, these percentiles were in the mid and low ranges, respectively.

In performing the MAAP calculations, the IPE team made the following assumptions:

- During the source-term analysis no credit was taken for accident mitigation activities because the back-end analysis assumed a 48-hour mission time (twice the IPE mission time of 24 hours) and yet found no early containment failures;
- Based on an evaluation of the VEGP containment/cavity design, the possibility of core debris leaving the cavity was assumed to be negligible;
- No failed equipment was recovered during the back-end analysis except that specifically defined in the front-end event trees;

- Accident sequences were evaluated by assuming that no more than two CCUs were operating; and

- No credit was taken for fission product retention in the auxiliary building.

The IPE team concluded that:

- None of the VEGP functional core damage sequences exceeded the source-term screening criteria used for determining reportability to the NRC; and

- The VEGP did not have any plant-specific vulnerabilities to severe accidents.

4. REFERENCES

1. Georgia Power Company. "Vogtle Electric Generating Plant Units 1 and 2: Individual Plant Examination Report in Response to Generic Letter 88-20." December 1992.

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3. Sherman, M. P., and M. Berman. "The Possibility of Local Detonation during Degraded Core Accidents in the Bellefonte Nuclear Plant." NUREG/CR-4803, SAND86-1180. Sandia National Laboratories. 1987.

4. Spencer, B. W., et al. "Results of EPRI/ANL DCH Investigations and Model Development." ANS/ENS Conference on Thermal Reactor Safety, Avignon, France. 1988.

5. IDCOR. "Key Phenomenological Models for Assessing Explosive Steam Generation Rates." Technical Report 14.1A. 1983.

6. Conax Corporation. "Design Qualification Material Test Report for Materials Used in Conax Electric Penetration Assemblies and Electric Conductor Seal Assemblies." IPS-325, rev. D. 1985.

7. Walker, J. V. "Reactor Safety Research Semi-Annual Report." NUREG/CR-5039. SAND87-2411. 1987.

8. Gabor, Kenton and Associates. "Recommended Sensitivity Analyses for an Individual Plant Examination Using MAAP 3.0B." EPRI TR-100167. 1991.

APPENDIX
IPE EVALUATION AND DATA SUMMARY SHEET

PWR Back-End Facts

Plant Name

Vogtle Units 1 and 2

Containment Type

Large, dry

Unique Containment Features

The VEGP cavity was expected to remain dry for most of the accident sequences.

The VEGP consists of two containment heat removal systems: eight containment cooling units (CCUs) and two containment spray system (CSS) trains. However, only the CCUs are capable of removing decay heat from the containment over the long term. The CSS can function only as a short-term containment pressure reduction system.

Unique Vessel Features

None

Number of Key Plant Damage States

14

Ultimate Containment Failure Pressure

112 psig (5th percentile value)

Additional Radionuclide Transport and Retention Structures

None

Conditional Probability that the Containment Is Not Isolated

0.004

Important Insights, Including Unique Safety Features

Listed under Unique Containment Features

Implemented Plant Improvements

No back-end-related plant improvements are reported in the submittal.

C-Matrix

Because no PDS contributes to more than one release category, the C-matrix would not provide any more information than appears in Table 4.7-7, page 4-48 of the submittal.