SCIE-NRC-238-95

# Callaway Unit 1
# Technical Evaluation Report
# on the
# Individual Plant Examination
# Back-End Analysis

H. A. Wagage
J. F. Meyer

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SCIENTECH, Inc.
11140 Rockville Pike, Suite 500
Rockville, Maryland 20852

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## E. EXECUTIVE SUMMARY

This technical evaluation report documents the results of the SCIENTECH review of the back-end portion of the Callaway Plant Unit 1 Individual Plant Examination submittal. [1, 2] The Callaway Plant is owed and operated by the Union Electric Company.

### E.1 Plant Characterization

The Callaway Plant is one of two units designed, built, and licensed under the Standardized Nuclear Unit Power Plant System concept. The sister unit is Wolf Creek Generating Station. Callaway is a 4-loop, Westinghouse-designed, pressurized water reactor (PWR) with a licensed core power level of 3565 MWt. The Nuclear Steam Supply System is housed in a large, dry containment structure designed by Bechtel Power Corporation. The containment walls and dome are reinforced, carbon-steel-lined concrete.

The median value of ultimate containment failure pressure of Callaway was calculated to be 135 psig.

### E.2 Licensee's IPE Process

The methodology used to perform the Callaway IPE consisted of a Level 2 probabilistic risk assessment (PRA) and internal flooding analysis using a small event tree/large fault tree approach. The containment analysis included the following:

- Identification of plant damage states that characterize the reactor, containment, and core-cooling systems at the start of core damage;
- Modeling of the thermal-hydraulic behavior of the reactor and containment and in- and ex-vessel fission product behavior for each of the plant damage states (PDSs);
- Characterization of containment failure modes;
- Selection of a containment event tree and its quantification for each plant damage state; and
- Characterization of radionuclide releases.

Development of the PRA at Callaway involved the participation of three groups : the Licensing and Fuels Safety Analysis and Reactor Design Group (SARD); the prime contractor, Halliburton NUS (NUS); and three subcontractors to SARD, i.e., Bechtel Power Corporation (BPC), Cermak Fletcher Associates, Inc. (CFA), and ~~KA.

The submittal notes the following with respect to the independent review performed of the PRA:

> The Callaway IPE did have independent review by virtue of the consultant, inter-departmental, Wolf Creek Nuclear Operating Corporation, and [CFA]. Comments from the reviews were generated during the performance of the IPE and were incorporated to complete the various tasks. Such comments and their resolutions are not included in this submittal, but have been retained by Union Electric either as part of the individual calculation packages or in separate files.

## E.3 Back-End Analysis

The IPE team used a containment event tree (CET) as a logic model to describe possible paths that an accident sequence might progress along, given an initial set of conditions defined by a PDS. The team selected the CET top events for the following purposes:

- To represent the uncertainties in physical phenomena (e.g., direct containment heating (DCH), containment loading);
- To assess operator recovery and mitigation actions; and
- To assess consequential failure of important systems, given the occurrence of specific physical phenomena (e.g., hydrogen burns) or as a result of the general severe accident environment.

The IPE team kept the number of top events in the CET low (a total of eight) and analyzed them by using decomposition event trees (DETs). The IPE team developed CETs for each PDS. The top events in the CETs consisted of phenomenological events or processes and consequential system failures resulting from physical phenomena or from accident environments considered important to the definition of the source term and the time, mode, and location of containment failure. Before selecting the CET top events, the IPE team reviewed the severe accident phenomena and containment events specified in NUREG-1335, the detailed set of events developed for NUREG-1150 and NUREG/CR-4551, and the results of past probabilistic safety analyses and the Industry Degraded Core Restoration (IDCOR) Program.

The time periods considered for the CET top events were:

- Before RV failure;
- At or within a few hours of the time of RV failure; and
- Late, i.e., many hours after RV failure.

The most dominant containment failure modes at Callaway were late overpressure leaks, either through the auxiliary building (30.1 % of the total CDF) or directly to the environment (15.1 %). The only other containment failure mode associated with more than 1 % of the core damage frequency was the long-term basemat melt-through (5.3 %). Source-term releases for this mode would be small and late. As a consequence of Callaway's independent and redundant design features, the containment isolation system had a low failure probability.

### E.4 Generic and Containment Performance Improvement Issues

In response to the staff's RAI, Union Electric gave a limited description of its response to the recommendations of the Containment Performance Improvement (CPI) Program:

> A detailed walkdown of the containment was carried out by Union Electric, NUS Corporation, and Gabor, Kenton, and Associates (GKA) personnel to ascertain the potential for localized hydrogen accumulation. The walkdown report indicates that the containment is relatively open and that no appreciable accumulation of hydrogen can take place and therefore no localized hydrogen combustion is expected.

The IPE team identified no plant or procedure improvements that should be made in response to the recommendations of the CPI Program. To address containment vulnerability to global hydrogen burn the licensee used CET top events.

### E.5 Vulnerabilities and Plant Improvements

In order to screen the IPE results for vulnerabilities, Union Electric employed the process described in NUMARC 91-04, "Severe Accident Closure Guidelines," (Section 3.4.2.2, page 3.4-15). The IPE team identified no back-end vulnerabilities at the Callaway plant, and recommended no back-end plant improvements.

### E.6 Observations

The most dominant containment failure modes at Callaway were late overpressure leaks, either through the auxiliary building (30.1 % of the total CDF) or directly to the environment (15.1 %). The only other containment failure mode associated with more than 1 % of the core damage frequency was the long-term basemat melt-through (5.3 %). Source-term releases for this mode would be small and late. As a consequence of Callaway's independent and redundant design features, the containment isolation system had a low failure probability.

The back-end portion of the submittal [1, 2] appeared to be complete and consistent with the guidelines in Generic Letter 88-20.

# 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 Callaway Plant Unit 1 Individual Plant Examination (IPE) submittal. [1, 2] This technical evaluation report complies with the requirements for 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 requested in the "Submittal Guidance Document," NUREG-1335;
- To help NRC staff assess the strengths and the weaknesses of the IPE submittal; and
- To complete the IPE Evaluation Data Summary Sheet.

Based in part on SCIENTECH's preliminary review of the Callaway IPE submittal, the NRC staff submitted a Request for Additional Information (RAI) to the Union Electric Company on July 18, 1995. The Union Electric Company responded to the RAI in a document dated September 28, 1995. [2] This final TER is based on the original submittal and the response to the RAI.

Section 2 of the TER summarizes our review findings and briefly describes the Callaway 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 3 presents our overall evaluation of the back-end portion of the Callaway IPE based on our submittal-only review. Section 3 also outlines the conclusions and insights gained, plant improvements identified, and utility commitments made as a result of the IPE. References are given in Section 4. The appendix contains an IPE evaluation and data summary sheet.

## 1.2 Plant Characterization

Callaway Plant is one of the two units designed, built, and licensed under the Standardized Nuclear Unit Power Plant System concept. The sister unit is Wolf Creek Generating Station. Callaway Plant is a 4-loop, Westinghouse-designed pressurized water reactor (PWR) with a licensed core power level of 3565 MWt. The Nuclear Steam Supply System is housed in a large, dry containment structure designed by Bechtel Power Corporation. The containment walls and dome are reinforced, carbon-steel-lined concrete.

The power block, which is the standardized portion of the plant, is comprised of the following major facilities (section 1.2, page 1-3):

- Reactor containment building. Houses the reactor, reactor coolant system, steam generators, pressurizer, reactor coolant pumps, accumulators, and the containment air coolers;

- Auxiliary building. Houses the engineered safety features (ESFs) and nuclear auxiliary systems equipment;

- Turbine building. Houses turbine generator, condensers, main feed pumps, and other power-conversion equipment;

- Fuel building. Houses the new fuel storage vault, the spent fuel handling system, and a portion of the spent fuel pool cooling and cleanup system;

- Radwaste building. Houses the radioactive waste treatment facilities and boron recycle system components;

- Control building. Houses the main control room , computers, the Class 1E switchgear, the Class 1E battery rooms, the access control area, cable spreading rooms, and the main control room habitability systems;

- Storage tanks. These include the condensate storage tank, the refueling water storage tank, the reactor makeup water storage tank, the refueling water storage tank, the reactor makeup water storage tank, the demineralized water tank, and the emergency fuel oil storage tanks;

- Diesel generator building. Houses the diesel generators and associated equipment; and

- Transformer vaults. Oil retaining structures for the main transformers, startup transformer, station service transformer, unit auxiliary transformer, and ESF transformers.

The nonstandardized portion of the plant (compared with the Wolf Creek plant) includes the following:

- Circulating water cooling tower;
- Ultimate heat sink retention pond and cooling tower;
- Essential service water pump house;
- Administrative and support buildings;
- Technical support center;
- Emergency operations facility;
- Simulator and training complexes;
- Switchyard and offsite power sources;

- Storage tanks; and
- Security facilities.

The dimensions of the Callaway containment are as follows:

| | |
|---|---|
| Internal diameter | 140 ft |
| Interior height | 205 ft |
| Height to spring line | 135 ft |
| Base slab thickness | 10 ft |
| Cylinder wall thickness | 4 ft |
| Dome thickness | 3 ft |
| Liner plate thickness | 0.25 in |
| Internal free volume | 2.5E6 $ft^3$ |

## 2. TECHNICAL REVIEW

### 2.1 Licensee's IPE Process

#### 2.1.1 Completeness and Methodology.

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

The methodology used to perform the Callaway IPE consisted of a Level 2 probabilistic risk assessment (PRA) and internal flooding analysis using a small event tree/large fault tree approach. The containment analysis included the following (Section 1.3, page 1-5):

- Identification of plant damage states that characterize the reactor, containment, and core-cooling systems at the start of core damage;

- Modeling of the thermal-hydraulic behavior of the reactor and containment and in- and ex-vessel fission product behavior for each of the plant damage states (PDSs);

- Characterization of containment failure modes;

- Selection of a containment event tree and its quantification for each plant damage state; and

- Characterization of radionuclide releases.

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

Multi-unit effects were not applicable to the single-unit Callaway plant.

The submittal notes that extensive design, configuration, and document control programs ensured that drawings and procedures used in the IPE reflected the as-built, as-operated plant (Section 2.4.4, page 2-13). In addition, for each record cited in the IPE calculation packages Union Electric did a cross-comparison with its current version, but found no differences between them.

The IPE team conducted two walkdowns as part of the containment performance analysis portion of the IPE (Section 2.4.5, page 2-14 of the submittal). The first walkdown was

conducted to obtain information to support auxiliary building modeling (nodalization) in the Callaway MAAP model, to determine if there were any submerged break locations, and to obtain information on fire suppression sprays. An employee of Union Electric and another from Gabor Kenton Associates, Inc. (GKA) conducted the first walkdown.

Three individuals, employed respectively by Union Electric, GKA, and NUS, conducted the second walkdown. This one was conducted in the containment building to address various MAAP modeling and containment performance issues. One of these issues related to the possibility of hydrogen pocketing, which could lead to local burn and threaten the integrity of the containment. (See Section 2.5 of this TER.)

### 2.1.3 Licensee Participation and Peer Review.

One of the major objectives of the Callaway Probabilistic Risk Assessment (PRA) was to make utility personnel knowledgeable about using PRA techniques. The utility plans to integrate PRA into its Nuclear Division management structures and to develop and implement an application plan. To accomplish this, Union Electric designated a group of individuals from among its permanent staff of engineers in the Licensing and Fuels Department to be responsible for developing and applying PRA techniques.

Development of the PRA at Callaway involved the participation of three groups (Section 5.1.1, page 5-1): the Licensing and Fuels Safety Analysis and Reactor Design Group (SARD); the prime contractor, Halliburton NUS (NUS); and three subcontractors to SARD, i.e., Bechtel Power Corporation (BPC), Cermak Fletcher Associates, Inc. (CFA), and GKA.

The Supervising Engineer of SARD was the Project Manager with overall responsibility for the Callaway IPE program. Because of the need for technology transfer, however, he received technical direction from the NUS Program Director and Program Manager. This technical direction was more important during the early phases of the PRA; as the effort proceeded, the Supervising Engineer took on an increasing role in managing the technical aspects of the work.

The Union Electric engineers participated in the following back-end tasks in the development of the Callaway PRA:

- MAAP Analysis. Union Electric performed all MAAP analysis including construction of the MAAP parameter file in support of the Level 1 and 2 analyses. GKA provided training and technical assistance;

- Plant Damage States. NUS developed and Union Electric reviewed the PDSs;

- Containment Ultimate Strength Analysis. BPC performed this task and Union Electric and NUS reviewed the results;

- Containment Event Tree/Decomposition Event Tree (CET/DET) Construction and Quantification. NUS performed this task; and

- Sensitivity/Uncertainty Analyses. NUS performed all Level 2 sensitivity analyses and Union Electric reviewed the results.

The Callaway IPE was performed in accordance with Union Electric Quality Assurance Procedure, FDP-ZZ-04002, Revision 1, "Performance of Callaway Individual Plant Examination (IPE)."

Section 5.3.2, page 5-10 of the submittal notes the following:

> The Callaway IPE did have independent review by virtue of the consultant, inter-departmental, Wolf Creek Nuclear Operating Corporation, and [CFA]. Comments from the reviews were generated during the performance of the IPE and were incorporated to complete the various tasks. Such comments and their resolutions are not included in this submittal, but have been retained by Union Electric either as part of the individual calculation packages or in separate files.

## 2.2 Containment Analysis

### 2.2.1 Front-end Back-end Dependencies.

The Callaway IPE team calculated the following core damage frequencies (CDFs):

- CDF, excluding internal flooding events: 4.064E-5 per year, and

- CDF, including internal flooding events: 5.846E-5 per year.

As described in Section 3.1.2.2.1, page 3.1-25 of the submittal, the analysts used two approaches to assess core damage. They assumed core damage would not occur in sequences where the required equipment operated to mitigate the consequences of the initiating event. They assumed core damage occurrence when it was obvious to them that a sequence would end in core damage. For those sequences where the analysts were unsure of the final result, they employed the MAAP computer code. In a MAAP analysis, core damage does not occur if:

- The calculated peak fuel ring segment clad temperature is below 2200°F;

- The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1% of the total amount of zircaloy in the reactor;

- The core remains coolable during and after a break in the primary system; and

- Core temperature is reduced and decay heat removal is established for an extended period of time.

Internal flooding was the largest contributor (30.48%) to the Callaway CDF; the contribution from station blackout was almost as large (30.30%). Other significant contributors were transient-induced reactor coolant pump seal loss of coolant accidents (LOCAs), intermediate and small LOCAs, and loss of all service water.

The entry points to the CETs were PDSs and the IPE team defined the PDS characteristics by selecting a set of key system operation-related parameters considered to be important to: (1) accident progression in the containment; (2) the time, mode, and location of containment failure; and (3) the radionuclide source term. The parameters used to define the PDSs included the functional status of important systems; state variables, which are determined by systems operation (e.g., Reactor Coolant System (RCS) pressure); accident initiator type; and timing of key events (e.g., power recovery).

The IPE team used nine parameters to define the Callaway PDSs. Table 1 lists these parameters and their relative contributions to the total CDF.

Using the same nine parameters, the team defined 81 PDSs for the following sequences. (See Figure 4.3-1 of the submittal):

- PDSs 1 and 2. Failure of containment isolation;
- PDSs 3 through 23: SBO; PDSs 3 through 18: Power recovery before reactor vessel (RV) failure; PDSs 19 through 22: Power recovery after RV failure; and PDS 23: No power recovery;
- PDSs 24 through 51. All transient initiated sequences other than SBO;
- PDSs 52 through 59. Large and intermediate break LOCAs;
- PDSs 60 through 79. Small break LOCAs;
- PDS 80. Containment bypass interfacing systems LOCA; and
- PDS 81. Containment bypass SGTR.

Of these 81 PDSs, 47 had a frequency (> 1E-10 per year) assigned to them. Table 4.3-1 of the submittal lists the 47 PDSs by frequency. The top five PDSs (PDSs 51, 23, 30, 19, and 79) comprised approximately 71% of the CDF. The remaining PDSs comprised the other 29%, with 99.8% of the CDF caused by the first 26 PDSs. The remaining PDSs contributed approximately 0.2% to the total CDF. These dominant PDSs are described as follows (Section 4.3.3, pages 4.3-13 and 4.3-14 of the submittal):

- PDS 51 (24.7% of the CDF). Represents non-SBO transient sequences with containment spray and containment heat removal unavailable, between 200-2000 psig of vessel pressure, and no injection sources;

- PDS 23 (17.5% of the CDF). Represents SBO sequences with no power recovery before containment failure, no containment spray, no containment heat removal, injection sources unavailable, and between 200-2000 psig of vessel pressure;

**Table 1. Callaway PDS Parameters and Their Contribution to CDF**

| PDS Parameter | Option | % CDF |
|---|---|---|
| CONBYPASS--Containment bypass status | No bypass | 98.1 |
| | Interfacing systems LOCA (ISL) | 0.3 |
| | Steam generator tube rupture (SGTR) | 1.6 |
| CONISOLAT--Containment isolation status | Not isolated | 0 |
| | Isolated | 100 |
| TRANLOCA--Transient or LOCA type sequence | Transient | 78.3[2] |
| | Large/intermediate LOCA | 6.3 |
| | Small LOCA | 13.5 |
| SBO--Station blackout sequence | Yes | 32.5[1] |
| | No | 45.8 |
| POWRECOV--Power recovery timing | Before reactor vessel failure | 11.1 |
| | Before containment failure | 35.2 |
| | No power recovery | 53.7 |
| RECSPRAYS--Containment recirculation sprays | Yes | 44.5[2] |
| | No | 53.6 |
| CNHEATREM--Containment heat removal | Yes | 44.7[2] |
| | No | 53.4 |
| RCSPRES--RCS pressure during core damage/at vessel failure | Hi-Hi (> 2500 psig) | 0.8[2] |
| | High (2000 - 2500 psig) | 1.3 |
| | Lo-Hi (200 - 2000 psig) | 89.7 |
| | Lo-Lo (< 200 psig) | 6.3 |
| INVESSINJ--Status of in-vessel injection | On | 9.1[2] |
| | Deadheaded | 4.2 |
| | Failed | 84.8 |

[1] This does not add up to 100 because no distinction was made for LOCA initiated sequences: a LOCA initiated sequence with total loss of AC power has a very low frequency and was not included in the Level 1 analysis.

[2] This does not add up to 100 because 1.9% of the CDF of containment bypass sequences was excluded.

- PDS 30 (13.5% of the CDF). Represents transient events with power available, containment spray and heat removal available via fan coolers, between 200-2000 psig of vessel pressure, and no injection system available;

- PDS 19 (8.23% of the CDF). Represents SBO sequences with power recovery after reactor vessel failure but before containment failure, containment spray and containment heat removal available, between 200-2000 psig of vessel pressure, and no injection system available; and

- PDS 79 (7.00% of the CDF). Represents small LOCA sequences (dominated by a seal LOCA) with power recovery after reactor vessel failure with containment spray and containment heat removal unavailable, between 200-2000 psig vessel pressure, and no injection system available.

The following is a summary of important results depicted in Table 1 above:

- Containment bypass sequences (SGTR and ISL) comprise 1.9% of the total CDF;

- Transient (non-SBO) and very small LOCA sequences contribute 45.8% of the CDF;

- SBO sequences contribute 32.5% of the CDF;

- Small LOCAs contribute 13.5% of the CDF;

- Large/intermediate LOCAs contribute the remaining 6.3%; and

- For all noncontainment bypass sequences, containment spray and containment heat removal are available in approximately 45% of the sequences.

2.2.2 Containment Event Tree Development.

The IPE team used a CET as a logic model to describe possible paths that an accident sequence might progress along, given an initial set of conditions defined by a PDS. The team selected the CET top events for the following purposes (Section 4.5, page 4.5-1 of the submittal):

- To represent the uncertainties in physical phenomena (e.g., direct containment heating (DCH), containment loading);

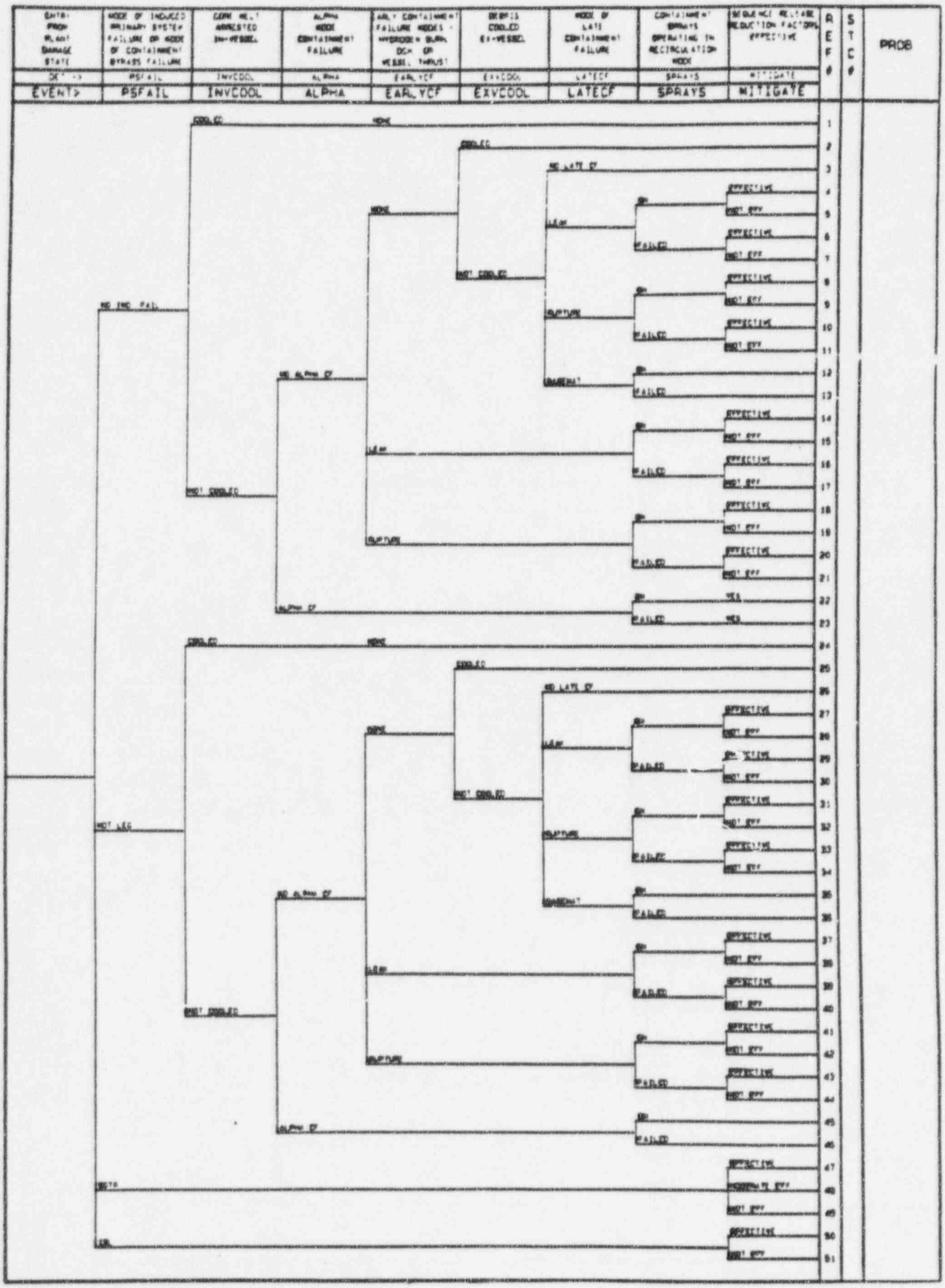
- To assess operator recovery and mitigation actions; and

- To assess consequential failure of important systems, given the occurrence of specific physical phenomena (e.g., hydrogen burns) or as a result of the general severe accident environment.

The IPE team kept the number of top events in the CET low (a total of eight) and analyzed them by using decomposition event trees (DETs). Figure 4.5.1 of the submittal shows the Callaway CET, which is reproduced in this report as Figure 1.

The IPE team developed CETs for each PDS. The top events in the CETs consisted of phenomenological events or processes and consequential system failures resulting from physical phenomena or from accident environments considered important to the definition of the source term and the time, mode, and location of containment failure. Before selecting the CET top events the IPE team reviewed the severe accident phenomena and containment events specified in NUREG-1335, the detailed set of events developed for NUREG-1150 and for NUREG/CR-4551, and the results of past probabilistic safety analyses and the Industry Degraded Core Restoration (IDCOR) Program.

**Figure 1. The Callaway containment event tree**



The time periods considered for the CET top events were:

- Before RV failure;
- At or within a few hours of the time of RV failure; and
- Late, i.e., many hours after RV failure.

The type of events considered in the CETs are listed in Table 4.5-1 of the submittal, which is reproduced as Table 2 of this report. Several of the events listed in Table 2 were combined under single event headings because they occurred nearly simultaneously and their effects were interrelated. For example, the events "Direct containment heating" and "Mode of early containment failure" were combined because they related to, or contributed to, overpressure loading of the containment at the time of vessel failure. Table 3 of this report lists all of the CET top events used and their branch possibilities.

2.2.3 Containment Failure Modes and Timing.

The IPE team investigated the following failure modes most likely to occur at the Callaway plant:

- Containment bypass;
- Failure to isolate;
- Vapor explosions;
- Combustion processes;
- Steam overpressurization;
- Core-concrete interaction (basemat melt-through);
- Blowdown forces (vessel thrust force);
- Liner melt-through (direct contact of containment shell with fuel debris);
- Failure of containment building penetrations; and
- Failure of active systems necessary to maintain containment isolation.

Bechtel Power Corporation (BPC), a contractor to Union Electric, calculated the Callaway containment failure pressure. Because the IPE team judged the probability of global detonations to be quite small, BPC investigated only the containment threat from static loads. BPC calculated a median ultimate containment pressure capacity of 134.9 psig (at 50% confidence level) and a lower-bound capacity of 126.9 psig (at 95% reliability with 50% confidence). These failure pressures represented the threshold beyond which yielding would occur of the liner, hoop pre-stress tendons, and hoop reinforcement.

Table 2. Potentially Important Containment Event Topics

| Phenomenological Events | Time Phase |
|---|---|
| Debris Cooled In-Vessel | Before RV Failure |
| In-vessel Steam Explosion | Before RV Failure |
| Mode/Time Vessel Failure | At/Near RV Failure |
| Direct Containment Heating | At/Near RV Failure |
| Early $H_2$ Burn/Detonation | At/Near RV Failure |
| Debris Dispersal Out of Cavity | At/Near RV Failure |
| Ex-vessel Steam Explosion/Spikes | At/Near RV Failure |
| Liner Melt-through | At/Near RV Failure |
| Mode of Early Containment Failure | At/Near RV Failure |
| Debris Cooled Ex-vessel | Longer Term |
| Late $H_2$ Burn/Detonation | Longer Term |
| Late Containment Over Pressure Failure | Longer Term |
| Mode of Late Containment Failure | Longer Term |
| Auxiliary Building Fission Product Mitigation Effectiveness | Longer Term |

| Operator, Recovery, Mitigation Actions | |
|---|---|
| In-vessel Injection Restored | Before/After RV Failure |
| RCS Depressurized | Before RV Failure |
| Power Recovery | Before/After RV Failure |
| Containment Spray Available | After RV Failure |
| Containment Heat Removal Available | After RV Failure |

| Consequential Systems Failures | |
|---|---|
| Late Spray Failure | After RV Failure |
| Late Containment Heat Removal Failure | After RV Failure |

**Table 3. Callaway CET Top Events and Their Branch Possibilities**

| CET Top Event | Branch Possibilities |
|---|---|
| Mode of induced primary system failure or containment bypass failure | 1. No induced RCS failure<br>2. Rupture of a hot leg (or the pressurizer surge line)<br>3. Initial or induced SGTRs<br>4. Interfacing system LOCA |
| Core melt arrested in vessel | 1. Debris cooled in vessel (no vessel failure)<br>2. Debris not cooled in vessel |
| ALPHA containment failure mode | 1. No ALPHA mode containment failure<br>2. ALPHA mode containment failure |
| Modes of early containment failure | 1. No early containment failure<br>2. Leak (approximately 14 sq. in.)<br>3. Rupture (approximately 144 sq. in.) |
| Debris cooled ex-vessel | 1. Debris cooled ex-vessel<br>2. Debris not cooled ex-vessel |
| Modes of late containment failure | 1. No late containment failure<br>2. Leak<br>3. Rupture<br>4. Basemat melt-through |
| Containment sprays operating in recirculation mode | 1. On<br>2. Failed |
| Release reduction factor effectiveness | 1. Effective<br>2. Moderately effective<br>3. Not effective |

In calculating the containment failure pressure, BPC identified the "weak links" in the containment structure. The weaknesses that were controlling based on their respective median and lower-bound capacities were:

- Hoop membrane stress failure as the result of the yielding of the liner, hoop tendons, and hoop reinforcement at the shell mid-height region; and

- Liner plate tearing around large pipe penetrations at the shell mid-height region.

Key components included in the containment structural analysis were the base slab/cylinder interface, cylinder mid-section, dome, liner plate, access openings, and pipe/electrical penetrations. BPC used finite element analysis for the containment strength evaluation.

BPC calculated containment failure pressure at 400°F. As an upper bound, BPC calculated the impact of temperatures of up to 800°F on the containment failure pressure. The IPE

team reevaluated the containment failure pressure for the weak link corresponding to a large pipe penetration at the mid-height region based on a containment temperature of 800°F (Section 4.4.2.7, page 4.4-32 of the submittal). The submittal reports that the containment strength, and associated failure probabilities, were not significantly affected by increased temperatures. The submittal notes that the Callaway electrical penetrations (EPAs) were treated to simulate an age of up to 40 years (the plant's licensed life) and then were qualified for a severe accident with a sustained maximum (containment) internal pressure of 60 psig at a sustained internal temperature of 340°F.

The IPE team reviewed the results of a test in which an EPA similar to those used at the Callaway Plant was equivalently aged up to 40 years and then assessed for its ability to withstand a severe accident. Identified as a "D.G. O'Brien" EPA, it survived a sustained maximum pressure of 140 psig at a sustained maximum temperature of 361°F without any observable leakage. The sealants (silicone) survived this temperature because the sealants located at the outer end of the nozzle and header plate were influenced by the ambient temperature outside the containment. In this test the header plate was located 10 inches away from the inside face of the containment. For the Callaway containment (concrete) this distance is 50 inches, and therefore, the sealant would be exposed to a lower ambient temperature at the same containment temperature. Based on this information the IPE team concluded that the internal pressure capacity of each Callaway containment EPA was at least 140 psig at an internal temperature of at least 400°F.

### 2.2.4 Containment Isolation Failure.

The Callaway containment pressure boundary includes penetrations for equipment and personnel hatches, piping and electrical penetration sleeves, the fuel transfer tube penetration sleeve, and the purge line penetration sleeves (Section 4.1.1, page 4.1-4 of the submittal).

The IPE team considered an extensive list of containment penetrations for possible isolation failures. During the initial screening they screened out the isolation failure sizes less than 1 inch in diameter arguing that a release through such a line would be too small for a sufficient release from the containment. This argument is consistent with those of several other IPEs; some IPE analysts chose even a larger limit of 2 inches. After the initial screening and evaluation a more detailed review was performed for each of the following penetrations (pages 3.2-110 through 3.2-113):

- Main steam lines;
- Main feedwater lines;
- Containment spray system;
- RHR system recirculation lines;
- Fuel pool cooling and cleanup system;
- Floor and equipment drainage system;
- Decontamination steam system;
- Containment hydrogen control system;

- Fire protection system;
- Breathing air supply system; and
- Containment purge system.

Of these only three penetrations (main steam lines, main feedwater lines, and containment purge system) were chosen to model in the containment isolation fault tree. Other penetrations were found not to be important because of various reasons. For example, reasoning for not modeling the breathing air supply system was given as follows (page 3.2-113):

> Penetration P-98 contains the line to the containment hose stations for the breathing air system. This is an air filled line and both isolation valves inside and outside containment are manual valves that are locked closed. They would remain closed during an accident scenario.

In order to make the trees tractable in size, the IPE team did not question the status of containment isolation in the CETs. The team fo·r·d that failure of containment isolation at Callaway was generally independent of the other systems modeled in the PRA, including AC and DC power. Thus the team members analyzed the failure of containment isolation separately from the CETs. By developing the isolation function failure cutsets from a separate fault tree analysis, they ensured that all existing dependencies were modeled consistent with the Level 1 system fault tree models. Mainly because of the relative independence of the containment isolation system the team calculated a total containment isolation failure frequency below the 1.0E-10 per year truncation level (Section 4.3.2, page 4.3-6). Therefore, the team placed all of the sequence frequencies in the isolated category.

2.2.5 System/Human Responses.

As listed in Table 4.5-1 of the submittal (Table 2 of this report) the IPE team addressed the following potentially important CET conditions involving operator, recovery, and mitigation actions:

- In-vessel injection restored (before/after RV failure);
- RCS depressurized (before RV failure);
- Power recovery (before/after RV failure);
- Containment spray available (after RV failure); and
- Containment heat removal available (after RV failure).

However, a discussion between the licensee and NRC and its contractor for reviewing human reliability analysis of the IPEs revealed that no back-end operator, recovery, and mitigation actions were included in the analysis. The only exception was AC power recovery for which generic data available on WCAP 10541 and NUREG-1032 were used (table 3.3.2-5, page 57 of 63, reference 3).

2.2.6 Radionuclide Release Categories and Characterization.

The end points of the CETs represent the outcomes of possible in-containment accident progression sequences. The Level 1 system information was passed to the containment evaluation in discrete PDSs. The IPE team developed release categories (or source term categories) by grouping accident sequences with similar characteristics. Table 4 in this report lists the accident sequence characteristics selected for use in the definition of the Callaway source-term release categories and the contribution of each to the total CDF.

The IPE team defined "early" containment failure as that occurring during the first day of the accident (within 24 hours). This definition is different from those used in many IPEs and the NUREG-1150 study, where the time is based on the occurrence of vessel breach. The NUREG-1150 study defined early containment failure as that occurring before or within a few minutes of vessel breach. However, this difference in definition would not have significantly affected the back-end results. (See the timing of key events for accident scenarios listed in Tables 4.6-5 through 4.6-8 of the submittal.)

**Table 4. Callaway Accident Sequence Characteristics Selected for Source Term Category Definitions and Their Contribution to CDF**

| Accident Sequence Characteristic | Option | % CDF |
|---|---|---|
| Time of release | Early (1 day or less) | 2.1 |
| | late (after 1 day) | 52.8 |
| | No release | 45.0 |
| Mode of containment failure | ISL | 0.33 |
| | ALPHA containment failure | 0.01 |
| | SGTR | 1.63 |
| | Rupture | < 0.01 (early)<br>0.05 (late) |
| | Leak | 0.16 (early)<br>47.52 (late) |
| | Basemat penetration | 5.28 |
| | No release | 45.01 |
| Sprays available and effective in fission product mitigation | Effective | 2.4 |
| | Not effective (or not available) | 52.3 |
| | No release or not questioned | 45.3 |
| Release reduction factors effective for fission product mitigation | Effective | 31.9 |
| | Moderately effective | 0.7 |
| | Not effective | 17.1 |
| | No release or not questioned | 50.3 |

As described in Section 7.2, page 7-2 of the submittal, 45% of the CDF resulting from the severe accident sequences at Callaway involved no containment failure, which meant virtually

no radionuclide release to the environment. The most dominant containment failure modes were late overpressure leaks through either the auxiliary building (30.1% of the CDF) or directly to the environment (15.0% of the CDF). The source terms for these late releases were found to be very small and were less than the PWR-4 release category releases from WASH-1400. The other containment failure mode associated with more than 1% of the core damage frequency was the long-term basemat melt-through (5.3% of the CDF) which involved very small late releases.

Section 3.4.1.5, page 3.3-14 of the submittal, compares the magnitudes of the Callaway source-term categories (STCs) (which involved early releases) with the PWR-4 release category in the WASH-1400 study. This comparison revealed that all of the 15 early STCs at Callaway had one or more release fractions that exceeded the PWR-4 release magnitude. These release categories are listed in Table 3.4.1-1 of the submittal; the release group fractions exceeding the PWR-4 release magnitude are in bold type.

## 2.3 Quantitative Assessment of Accident Progression and Containment Behavior

### 2.3.1 Severe Accident Progression.

The IPE team performed MAAP calculations to determine the range and variation in containment response to be expected for a variety of accident scenarios. Other calculations were performed as sensitivity studies, primarily to gain general insights. Also, the IPE team performed MAAP calculations to determine release fractions for various source-term categories. These source-term cases were run past containment failure and as such represented the entire accident sequence from initiating event to release completion. The CET results of severe accident progression are described below (see also Section 4.6.5, page 4.6-27 of the submittal).

*PDS 51*

PDS 51 was the dominant PDS (24.7 % of the total CDF) for the Level 1 accident sequences. This PDS was dominated (76 %) by flooding in the control building basemat, FL2, which led to the total loss of service water and the subsequent loss of injection, containment sprays, and containment heat removal.

Because of the relatively low (LO-HI) RCS pressure expected, no induced primary system failures were expected and the probability of an in-vessel steam explosion failing the vessel and the containment was at the low end of the range (0.0001). Cooling the debris in vessel was not possible because no injection was available. Due to the reduced RCS pressure, the likelihood of early containment failure by DCH or vessel thrust was negligible. The only likely mode of early containment failure was a residual contribution (0.002% of the total CDF) resulting from hydrogen burns.

Ex-vessel cooling was not available, either. Also, since containment sprays and containment heat removal were failed, long-term containment failure was a foregone conclusion, either as the result of overpressure or basemat melt-through.

*PDS 23*

PDS 23 consisted of 17.5 % of the Level 1 accident sequence frequency and contained SBO sequences with no AC power recovery. The accident progression of this CET was similar to that of PDS 51 described above.

*PDS 30*

PDS 30 contributed to 13.5% of the Level 1 accident sequence frequency and contained transient sequences with failure of all injection. Containment sprays and containment heat removal via the containment fan coolers were available. For early containment failures and for ex-vessel debris cooling in the sequences that did not undergo early containment failure the accident progression would be similar to that of PDS 51. Because containment sprays and containment fan coolers were available, it was very likely that the debris would be cooled ex-vessel (97.7 %). For the small fraction that was not cooled ex-vessel because of the availability of containment heat removal and sprays, no containment failure was predicted to occur.

*PDS 19*

PDS 19 contributed to 8.2% of the Level 1 accident sequence frequency and contained SBO sequences with recovery of AC power before containment failure. For the early containment failure modes, and up until the point where late containment failure was possible in the sequences that were not cooled ex-vessel, the accident progression was the same as for PDS 30. In PDS 19, the containment sprays began to operate upon restoration of power, and thus deinertion of the containment occurred. The probability of the subsequent hydrogen burning and the resultant pressure rise leading to containment failure was judged to be highly unlikely (0.01). If the containment did not fail by hydrogen burn, because containment heat removal and sprays were available after AC power recovery, no containment failure was predicted to occur.

*PDS 79*

PDS 79 contributed to 7 % of the Level 1 accident sequence frequency and contained small LOCA sequences with failure of all injection, containment sprays, and containment heat removal. The accident progression for this PDS was similar to that of PDS 51.

### 2.3.2 Dominant Contributors: Consistency with IPE Insights.

In Tables 6 and 7 of this report dominant contributors to Callaway containment failure are compared with those contributors identified during individual plant examinations performed at similar plants, and with the NUREG-1150 PRA results obtained at Zion.

The most dominant containment failure modes at Callaway were late overpressure leaks, either through the auxiliary building (30.1 % of the total CDF) or directly to the environment (15.1 %). The only other containment failure mode associated with more than 1 % of the core damage frequency was the long-term basemat melt-through (5.3 %). Source-term releases for this mode would be small and late. As a consequence of Callaway's independent and redundant design features, the containment isolation system had a low failure probability.

### 2.3.3 Characterization of Containment Performance.

The IPE team used decomposition event trees to analyze the top events of the Callaway CET for severe accident progression. These DETs are described briefly as follows (Section 4.6.3 of the submittal):

- Induced primary system failure. The IPE team used the assumptions of the NUREG-1150 In-vessel Expert Panel;

- Core melt arrested in-vessel. The IPE team used the assumptions in NUREG/CR-4551;

- ALPHA mode containment failure. The IPE team maintained that the values used in NUREG/CR-4551 for the probability of ALPHA mode containment failure (0.008 for low pressure and 0.0008 for elevated pressures) were overly conservative and substituted numbers that were 8 times less in value;

- Early containment failure. The IPE team considered early containment failure resulting from hydrogen burning, direct containment heating (DCH), and vessel thrust forces. MAAP calculations for an SBO sequence with no auxiliary feedwater and no power recovery showed that with a hydrogen burn and maximum DCH occurring the containment pressure would rise to a maximum of 100 psia after vessel failure. This value was well below the pressure at which containment failure mode was expected to occur. To account for uncertainty in the modeling process, the team used a conservative value of 1% as the likelihood of containment failure due to early hydrogen burns when a high concentration of hydrogen was present; the team used a value of 0.1 % otherwise;

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 IPE | 4.9E-5 | 0 | 0 | 3.4 | 0.4 | 96.2 |
| Wolf Creek IPE | 4.2E-5 | 0.11 | 3.8 | 0.2 | 0.14 | 95.8 |
| Callaway IPE | 5.8E-5 | 0.2 | 52.8 | 2.0 | 0 | 45.0 |

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

- Debris cooled ex-vessel. In sequences where water was supplied to the debris, the following conditions were considered. For deep pools it was indeterminate whether the debris pools were coolable [P(cooled) = 0.5]. For shallow debris pools it was likely that the debris was coolable [P(cooled) = 0.9]. For very shallow debris pools it was judged that, given the supply of water, the debris would be cooled;

- Late containment failure. If the containment had not failed in the earlier phases, the team used this DET to determine whether late containment failure could occur due to late hydrogen burning, long-term overpressure, or basemat melt-through. The IPE team performed several bounding calculations with power to assess the potential for failing the containment as the result of a late hydrogen burn. After comparing the maximum containment pressure with that calculated for compound fragility curves during the containment ultimate strength analysis, the IPE team concluded that it was highly unlikely that the containment would fail due to a late hydrogen burn [P(late hydrogen burn fails containment) = 0.01];

**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 |
| Callaway IPE | 5.8E-5 | 0.2 | 52.8 | 2.0 | 0 | 45.0 |

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

- Containment spray operation in recirculation. If any containment failure mode (leak, rupture, basemat melt-through, or ALPHA) was expected to occur in a sequence, the team used this DET to determine whether or not the containment sprays were operable at the time of the release. The operation of sprays would result in fission product scrubbing from the containment before any fission product release, thereby reducing the source term associated with the release; and

- Release reduction factors effectiveness. If the containment was predicted to fail by leakage or rupture, either early or late, or if the sequence was a bypass sequence, the team used this DET to determine the effectiveness of various release mitigation factors based on the sequence type.

### 2.3.4 Impact on Equipment Behavior.

The IPE team evaluated the ability of the containment fan coolers to function effectively in a post-core melt atmosphere (Section 4.8, page 4.8-4). The submittal notes that, although the four containment fan coolers at Callaway were not designed for severe core-melt accidents, they were designed to withstand design basis accidents and are safety-grade. Only two fan coolers are needed to remove decay heat from the containment during a severe accident, so there is considerable overcapacity. If containment sprays and residual heat removal are

available, then the fan coolers are not required. The IPE team concluded that fan cooler survivability during severe accidents was not at issue in assessing the Callaway containment performance results.

2.3.5 Uncertainty and Sensitivity Analyses.

The IPE team performed sensitivity analyses regarding the following:

- ALPHA mode failure probability;
- Induced hot leg and SGTR failures in Hi-Hi and high-pressure sequences;
- Core-melt arrest in-vessel;
- Early containment failure due to hydrogen burning;
- Direct containment heating;
- Debris bed coolability; and
- Fan cooler survivability.

Using the above sensitivity analyses, the IPE team concluded that the judgments and uncertainties regarding the Level 2 issues were not likely to have had a significant impact on the overall release magnitude or total core damage frequency. The dominance of the ISL release, STC 1, was most critical to the overall release results.

## 2.4 Reducing Probability of Core Damage or Fission Product Release

2.4.1 Definition of Vulnerability.

In order to screen the IPE results for vulnerabilities, Union Electric employed the process described in NUMARC 91-04, "Severe Accident Closure Guidelines." (section 3.4.2.2, page 3.4-15) The IPE team identified no back-end vulnerabilities at the Callaway plant. (section 3.4.2.4, page 3.4-17)

2.4.2 Plant Improvements.

No back-end plant improvements were identified during the IPE.

## 2.5 Responses to CPI Program Recommendations

As a result of the Containment Performance Improvement (CPI) Program, licensees are to consider certain recommendations as part of the IPE process. These recommendations, which are identified in Generic Letter 88-20, Supplement 3, apply to the Callaway plant as follows:

> Licensees with dry containments are expected to evaluate containment and equipment vulnerabilities to localized hydrogen combustion and the need for improvements (including accident management procedures) as part of the IPE.

In answer to the staff's RAI, Union Electric gave a limited description of its response to the recommendations of the CPI Program (page 47):

> A detailed walkdown of the containment was carried out by Union Electric, NUS Corporation, and Gabor, Kenton, and Associates (GKA) personnel to ascertain the potential for localized hydrogen accumulation. The walkdown report indicates that the containment is relatively open and that no appreciable accumulation of hydrogen can take place and therefore no localized hydrogen combustion is expected.

Union Electric addressed containment vulnerability to global hydrogen burn by using CET top events.

### 2.6 IPE Insights, Improvements, and Commitments

The IPE team identified the following unique safety features that would mitigate the effects of a large mass and energy release into the containment (Section 6.1, page 6-1 of the submittal):

- The free volume of the containment is in excess of 2.5 million cubic feet;

  Based on Bechtel's ultimate strength analysis of the Callaway containment, it can withstand pressure of roughly twice the design pressure; and

- The Callaway reactor cavity is designed to facilitate ex-vessel corium cooling and to minimize the probability of direct containment heating.

During the IPE the licensee identified no back-end plant improvements and made no commitments.

## 3. CONTRACTOR OBSERVATIONS AND CONCLUSIONS

The most dominant containment failure modes identified at Callaway were late overpressure leaks, either through the auxiliary building (30.1% of the total CDF) or directly to the environment (15.1%). The only other containment failure mode associated with more than 1% of the core damage frequency was long-term basemat melt-through (5.3%). Source-term releases for this mode would be small and late. As a consequence of Callaway's independent and redundant design features, the containment isolation system had a low failure probability.

In order to screen the IPE results for vulnerabilities, Union Electric employed the process described in NUMARC 91-04, "Severe Accident Closure Guidelines." In answer to the staff's RAI, Union Electric gave a limited description of its response to the recommendations of the CPI Program. The IPE team identified no back-end vulnerabilities at the Callaway plant and recommended no back-end plant improvements, either.

The IPE team identified the following unique safety features that would mitigate the effects of a large mass and energy release into the containment:

- The free volume of the containment is in excess of 2.5 million cubic feet;
- Based on Bechtel's ultimate strength analysis of the Callaway containment, it can withstand pressure of roughly twice the design pressure; and
- The Callaway reactor cavity is designed to facilitate ex-vessel corium cooling and to minimize the probability of direct containment heating.

SCIENTECH noted that the Callaway Plant IPE appears to meet the intent of Generic Letter 88-20. The submittal shows an understanding of accident progression, phenomenology, conservatisms, containment response, and radiological source terms.

## 4. REFERENCES


1. Union Electric Company. "Callaway Plant IPE, Final Report." September 1992.

2. Union Electric Company. "Responses to NRC RAI on Callaway IPE." September 28, 1995.

3. Communication between SCIENTECH and P. Swanson of Concord Associates. December 13, 1995.

Appendix
IPE Evaluation and Data Summary Sheet

PWR Back-End Facts

Plant Name

Callaway Unit 1

Containment Type

Large, dry

Unique Containment Features

The Callaway reactor cavity design facilitates ex-vessel corium cooling and minimizes the probability of direct containment heating.

Unique Vessel Features

None

Number of Plant Damage States

47

Ultimate Containment Failure Pressure

135 psig (median or 50th percentile value)

Additional Radionuclide Transport and Retention Structures

Mitigation of release through the auxiliary building was credited.

Conditional Probability that the Containment Is Not Isolated

Less than 1E-[illegible]

**Appendix (continued)**
**IPE Evaluation and Data Summary Sheet**

Important Insights, Including Unique Safety Features

See Unique Containment Features, above

Implemented Plant Improvements

No back-end plant improvements were considered.

C-Matrix

A C-Matrix can be generated from the information provided in Table 4.7.2-2 and Figure 4.7.2-1 of the submittal.

APPENDIX C

CALLAWAY PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(HUMAN RELIABILITY ANALYSIS)