ENCLOSURE 3

SCIE-NRC-207-91

TURKEY POINT IPE BACK-END AUDIT

Task 3

TECHNICAL EVALUATION REPORT

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Rev. 1

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1. INTRODUCTION AND SUMMARY

1.1 INTRODUCTORY COMMENTS

This final report sets out our technical evaluation of the Back-End portion of the Turkey Point Individual Plant Examination (IPE) performed by Florida Power & Light (FP&L). The audit carried out by SCIENTECH, Inc., and Energy Research, Inc., (ERI) was divided into three parts: initial review of the submittal document[1] in preparation for a site visit (Task 1); the site visit itself (Task 2); and the followup assessment of the Turkey Point IPE, based on the original submittal, the site visit, and FP&L responses to questions from NRC (Task 3).

On November 5, 1991, NRC received the Task 1 report, "Preliminary Evaluation of Turkey Point Individual Plant Examination (IPE) Back-End Submittal." In parallel with this effort, SCIENTECH/ERI helped NRC to formulate Round 1 questions to which FP&L distributed formal responses during the site visit of November 19-21, 1991. (See Section 2 of this report for a description of the visit.) During the site visit, a number of concerns that were raised in the Task 1 report were resolved, based on the information presented by FP&L and contained in the Round 1 responses[2]. However, the resolution of each of these issues was based for the most part on oral presentations. Other issues were left open.

Subsequently, the plan was to codify those issues that had been resolved orally by obtaining formal responses to Round 2 questions developed by NRC staff and contractors. Suggested questions for this second round were forwarded by SCIENTECH/ERI to NRC on November 25, 1991. Additional questions aimed at issue clarification were formulated and submitted to FP&L in a conference call from NRC on December 19, 1991. Because of the unavailability of key FP&L personnel, the questions aimed at clarification were not answered. The NRC staff decided to fold these latter queries into the Round 2 questions, which had not yet been issued.

Thus, this Task 3 report does not reflect any benefit that might have derived from SCIENTECH/ERI reviewing and evaluating the formal FP&L responses to the Round 2 questions. Instead, this report is based on what was contained in the original submittal, in the formal Round 1 responses, in materials handed out during the site visit, but not necessarily on what was discussed informally during the site visit. It is SCIENTECH/ERI's view that, eventually, the FP&L responses to the Round 2 questions will document and substantiate much of what has been resolved orally. However, for purposes here, the issues remain unresolved.

1.2 SUMMARY OF TECHNICAL EVALUATION REPORT

The presentation of the Back-End materials in the utility submittal is sound and relatively easy to evaluate. FP&L should be commended for presenting a Back-End structure that can be evaluated in a relatively straightforward manner. Gaps, open issues, and confusing sections do exist, but, on the whole, the submittal meets the spirit of what was requested in NUREG-1335 and the Generic Letter (88-20 and supplements[3, 4, 5, 6]).

Other than in the areas summarized below, the Turkey Point IPE submittal (Back-End) is technically sound, consistent with the level of effort appropriate for this activity. Insights from PRAs performed previously were used to guide development, both of the IPE we evaluated as well as our own performance as auditors. The NUREG-1150 study[7] was one of the sources. NUREG-1150 results and EPRI recommendations on how to use the MAAP computer code[8] were also used extensively in the submittal. In Section 3.1.3 of this report, specific comparisons are drawn between the Turkey Point plant and the similar Westinghouse PWRs with their large, dry containment buildings. The major difference between Turkey Point and Surry, is the former's relatively high probability of late containment failure. The Turkey Point late containment failure probability is 60 percent. As discussed in Section 3.1.3, a portion of this late failure may be due to an artifact of the methodology.

The site visit demonstrated that much more analysis has been done, and that FP&L staff understanding of severe accidents is considerably greater than reflected in the original submittal. This has a particular bearing on questions raised about two subjects: (1) hydrogen distribution and combustion and the ensuing challenges to the containment, and (2) the issues and systems involved in the Interfacing Systems Loss-of-Cooling Accident (ISLOCA) and unisolated steam generator tube rupture (SGTR). However, questions about these two areas must remain open until NRC receives formal responses that document their resolution by FP&L.

The key issues about which questions remain unresolved are the following:

- Hydrogen and the challenge of combustion to containment integrity
- Containment vulnerabilities and the potential for containment performance improvements (CPIs)
- Frequency and source term of the ISLOCA and the unisolated SGTR

- Justification and clarification of values used to develop the radiological source terms.

These issues are described in detail in Section 3.2. Questions about them are to be sent to FP&L.

The structure of the report follows that required in the Task Order. Thus this introduction is followed by a summary of the site visit audit in Section 2, and by the SCIENTECH/ERI findings in Section 3. The references appear in Section 4 and the Appendix contains the IPE evaluation and data summary sheet.

2. CONTRACTOR AUDIT

2.1 INFORMATION AUDITED AT THE SITE

During the site visit, which took place November 19, 20, and 21, 1991, members of the FP&L staff were asked the questions that appear in the SCIENTECH/ERI Task 1 report, Sections 4 and 5. Information audited at the site pertained to the following subject categories:

- CPI
- Role of uncertainties and "conservative" calculations
- Source terms
- Hydrogen burn
- ISLOCA
- Containment Isolation
- Containment event trees (CETs)
- Containment failure calculations
- Pressure and temperature histories

The discussion below is topical, consistent with the above list. Some general findings are presented first.

2.1.1 General Findings

Staffing and Level of Effort. FP&L employs five to six persons, who are fully dedicated to the FP&L IPE program at Turkey Point and St. Lucie. These same people will do the IPEEE for all units, which demonstrates the FP&L commitment to the IPE program. One concern, however, is how the Turkey Point operations staff itself can absorb and assimilate all that has been developed by the headquarter's staff located at the Juno Beach FP&L Office. The contractor is SAIC, with a fixed-price, $1.2M contract. The peer review was performed by Aaron Engineering. FP&L estimated the total level of effort to be about twice the contractor effort, or $2.4M. Only one SAIC employee was present for the duration of the site visit meetings. (SAIC's expert in Human Reliability)

2.1.2 Specific Items

2.1.1.1 Containment Performance Improvements

The original submittal did not recognize the potential for containment performance improvement (CPI). Generic Letter 88-20, Supplement 3[6], and the references in SECY-91-084[9] were discussed with members of the FP&L staff, who will consider adding CPI to the submittal modification.

2.1.1.2 Role of Uncertainties And "Conservative" Calculations

Direct Containment Heating (DCH) sensitivity assessments were discussed as examples of the determination of uncertainty. MAAP base cases were run with best-estimate values, including 0.03 dispersal. (This value is recommended by EPRI in their report on MAAP[8], page A-2, item 14.) Then other cases were run. (Note evidence of this on page 4-169 of the submittal.) For example, a sensitivity study was done with the value at 1.0 for the SBO seal LOCA.

The FP&L staff described the reactor cavity area and indicated the location of the sump pumps and instrument tube passageway, which is conducive to flooding.

According to the FP&L staff, the situation that presents the greatest uncertainty is in-vessel cooling of a damaged core when the cavity is flooded, but coolant is not recovered in-vessel. The assumption is that the vessel would always fail.

For all plant damage states, a 50-percent probability is assumed that a coolable debris bed is not formed in-vessel (PRCOOLDBV). The rationale for this percentile, according to the FP&L staff, is that PRCOOLDBV is equally indeterminate for all PDSs.

In the submittal it is considered conservative to include release paths through the steam generator in the small-break LOCA bins. (Note page 4-34 of the submittal.) Such action is considered conservative because (1) it takes a longer time to cause core damage with SGTR and thereby more time is available for prevention or mitigative actions and (2) the Emergency Operating Procedures (EOPs) call for SG isolation.

The site-visit discussions were helpful in better understanding the sensitivity analyses performed and the justification for some of the "conservative" assumptions that were made by FP&L.

2.1.1.3. Source Terms

During the site visit, source-term issues centered on the points raised on pages 11-13 of SCIENTECH/ERI's Task 1 report. FP&L claimed no double-counting, as the values used are based on specific MAAP analysis. NUREG-1150 values were not used.

FP&L stated no adequate reason for using zero for FCOR for Te -- only that MAAP assumed it. (Note page 4-210 of the submittal.)

As to why certain source-term groups were dropped, FP&L said that MAAP does up to 12 groups. In this Turkey Point analysis, seven groups were dropped after SAIC said their bearing on the study would be negligible.

Discussions at the site were informative. However, some of the values used and approaches taken by FP&L personnel could not be justified.

2.1.1.4. Hydrogen burn

Although not adequately cited in the submittal, sensitivity analyses were performed in an attempt to bound the hydrogen burn issue. For example, in the case of high-pressure vessel failure, FP&L tried to bound the hydrogen burn pressurization in the following ways: by (1) producing a maximum amount, which meant delaying core melt and vessel failure, changing the melt temperature (3,100K) and latent heat (5,000 BTU/lbm), and (2) forcing a burn by lowering the auto ignition temperature to 400K. (See page 4-170 of the submittal, MAAP runs T1IIIH10, 11, 12.) These MAAP runs did not fail the containment. FP&L also performed hand calculations of hydrogen burns. No sensitivity analysis was performed on hydrogen generation from steel oxidation.

According to FP&L, local hydrogen concentrations are not a problem because (1) steam inerts, (2) steam drives the mixing, resulting in the dilution of any concentrations, and (3) the Turkey Point containment is a relatively open one.
Ed Chow of the NRC staff asked if FP&L had sought out local concentrations. Staff members said that they had not. Even if they had local burns, they claimed, there was no important equipment in those locations.

FP&L staff members said the utility had considered hydrogen ignition sources other than those from AC power. Note that MAAP auto ignition is 983K. FP&L used 400K in a sensitivity analysis.

According to FP&L, the sensitivity analyses did not result in any containment failures.

The site-visit discussions showed that much more has been done in this area than reflected in the submittal. It would strengthen the IPE considerably if these assessments were formally described in an IPE modification.

2.1.1.5 ISLOCA

Far more analysis and assessment have gone into the ISLOCA evaluation than the IPE submittal indicated. This weakness might be compensated when the submittal is modified. NRC and SCIENTECH viewed drawings that showed the possible ISLOCA routes, the most probable one leading to the RWST. The RWST capacity (shared between units 3 and 4) allows for about 8 hours of injection, even though one tank has failed via ISLOCA. Because an operator action is required, FP&L did not take any credit for shared capacity. ISLOCA is a good candidate for accident management.

Note that FP&L personnel were not sure if the 4.4 percent contribution to core damage from ISLOCAs was from the dominant contributor, namely, from failure at the refueling water storage tank (RWST), or from the whole set.

As in the case of the hydrogen, above, the site-visit discussions revealed that much more has been done in this area than is reflected in the IPE submittal. However, the ISLOCA is potentially such an important issue that even more needs to be done, especially to characterize the ISLOCA source term, which probably meets the IPE generic letter screening criteria. (Note the further discussion of this matter in Section 3.1.2.1.)

2.1.1.6 Containment Isolation

In the containment isolation analysis, all cut sets were looked at. Purge was not included. The purge lines are isolated automatically upon signal. The only operator role is to verify closure. Thus the probability that the purge lines were open was considered so small it did not enter into the formulation of containment isolation probability.

The site-visit discussions dispelled the concerns raised after reviewing the FP&L responses to Round 1 (Back-End) questions 4 and 5. The response to question 4 gives the impression that a manual action is necessary in order to isolate the containment when in the purge mode. However, the containment is automatically isolated upon receipt of various signals. The only manual operation is a check that the containment has, in fact, been isolated.

2.1.1.7 Containment Event Trees (CETs)

CETs were quantified for the most part from NUREG-1150 results. Sensitivity analyses were performed using MAAP. (MAAP was also used to determine the containment pressure loadings and failure times.)

The CET displayed on page 4-77 of the submittal was not used in any of the analyses.

FP&L stated that some of the fault trees (FTs) were incomplete. (Note, for example, page 4-84 of the submittal.) FP&L staff said that, if modification is made to the submittal, the FTs will be completed. They said that this incompleteness does not affect the results.

The site-visit discussions clarified these previously open issues.

2.1.1.8 Containment Failure Calculations

Some ambiguity surrounds the FP&L calculation of containment failure. What FP&L did in fact is not consistent with what they claimed they did in the submittal. (The methodology described in Figures 4.6-2, 4.6-4, and 4.6-5 was not used.) Instead, a simpler approach was taken. Point values of MAAP-generated containment pressures were compared against probabilities (as a function of pressure) of the containment failing.

When FP&L personnel used NUREG-1150 Surry data in their analysis, they used the 20/21 scaling (ratio of the Turkey Point containment volume-to-power ratio to the Surry containment volume-to-power ratio). SCIENTECH/ERI believes that a better scaling is 22/19, derived from the ratio of Turkey Point to Surry fuel and cladding masses. FP&L agreed that the 22/19 scaling was more accurate, but asserted that their analysis was sound nevertheless.

Regarding failure via penetrations, they noted that all of the penetrations are below the 58-foot level, except those for the steam piping. Thus, high temperatures, high in the containment building, would not affect penetrations.

It remains unclear how FP&L determined if a containment failure was characterized as a rupture or a leak. According to FP&L, if MAAP calculates a failure at, say, 26 hours, then the pressure at 24 hours is compared to the failure probability curve and the probability is plugged into the FT/CET. Early failures are all caused by ruptures. The causes of late failures are evenly divided between leak and rupture. Note that the MAAP hole size is constant.

Asked why there is a relatively high probability of late containment failure with no vessel breach for a number of the plant damage states, FP&L responded that containment cooling has failed -- a generic result according to EPRI guidance.

There are no containment basemat failures. The maximum penetration is 2 feet.

The site-visit discussions left some matters unresolved. Clarifications have been requested.

2.1.1.9 Pressure and Temperature Histories

FP&L provided a set of MAAP-generated histories for SCIENTECH/ERI consideration. A summary of the site visit with suggested followup questions was provided to NRC on November 25, 1991. For the most part, these questions are duplicated here in Section 3.2.2.

2.2 PERSONNEL INTERVIEWED

During the Back-End portion of the site audit, the FP&L personnel interviewed were Ching N. Guey of the Reliability and Risk Assessment Group (Juno Beach) and Jay N. Kabadi of the Nuclear Fuels Group (Juno Beach). Both individuals seemed knowledgeable and capable.

2.3 WALKDOWNS

During the site visit, both units were operating, precluding entry into the containment buildings, and limiting the Back-End walkdowns. Most systems and structures of interest were inaccessible. Bill Milstead of the NRC staff and Jim Meyer of SCIENTECH did walk down the auxiliary building systems and the piping routes of the RWST for the possible ISLOCA.

3. CONTRACTOR FINDINGS REVIEW

3.1 REVIEW OF IPE SUBMITTAL

3.1.1 Review of The Plant and Containment Design Features That Contribute to The Progression of Severe Accidents

Table 1 summarizes the Turkey Point plant and containment system design features that could contribute to core melt progression and containment system response. Also listed are the design data for Surry (PWR with a subatmospheric containment) and Zion (PWR with a large dry containment), two of the reference plants used for the recently published NUREG-1150 risk study[7].

Containment pressure capacity is one of the most important attributes with direct impact on severe accident mitigation. The Turkey Point containment (Units 3 and 4) is constructed from prestressed concrete with a steel liner (for leak tightness). Comparisons of Turkey Point, Surry, and Zion containment capacities are enumerated in Table 2.

The following observations were made based on comparisons of the design features listed in Table 1:

- The Reactor Coolant System (RCS) for Turkey Point is similar to those of the other two Westinghouse PWRs (similar RCS volume-to-power ratios). This indicates that the reactor coolant system time windows (e.g., time to reach uncovery, boil-off) during severe accidents is similar for the three plants.

- The ratio of total fuel and zirconium mass to containment free volume as shown in Table 1 is about 15 to 40 percent larger at Turkey Point than at the Zion and Surry plants. This ratio indicates the potential severity of High Pressure Melt Ejection (HPME)/Direct Containment Heating (DCH), given similar cavity and containment configurations. The cavity and the shape of the keyway influence the potential for debris being trapped in their passages, through the cavity up into the upper containment compartments.

- The Turkey Point lower cavity and keyway configuration appears to be somewhere between Surry (nondispersive) and Zion (dispersive), in terms of its ability to trap core debris following RPV failure at high pressures. Therefore, a direct scaling of Surry pressurization loads to Turkey Point, based on a ratio of debris mass to containment free volume, is a good (but not conservative) indicator of the Turkey Point containment vulnerability to HPME/DCH pressurization.

TABLE 1 Comparison of Turkey Point, Zion, and Surry Plant And Containment Design Features That Contribute to The Progression of Severe Accidents

Feature	Turkey Point	Zion	Surry
Power Level, MW(t)	2,200	3,250	2,441
Volume of RCS Water, M3 Mass of Fuel, Tons Mass of Zicalloy, Ton Containment Volume, M3	261 79 19.4 43,891	360 98.5 20 77,070	260 80 16.5 50,970
RCS Water Volume/Power, M3/MW(t)	0.12	0.11	0.11
Containment Volume/Power, M3/MW(t)	20	24	21
Zr Mass/Containment Volume, Kg/M3	0.4	0.3	0.3
Fuel Mass/Containment Volume Kg/M3	1.8	1.3	1.6
Maximum H2 Generation from Zr Oxidation, Kg	859	886	723
H2 Generation from Fe Based on 20% Fe Oxidation, Kg	Not Known	91	156
Total H2 Generation, Kg	>859	977	879
Power Specific Hydrogen Generation, Kg/MW(t)	>0.4	0.3	0.4
Maximum Hydrogen Concent. in Containment, Moles/M3	>0.01	0.006	0.008
Containment Dispersive Characteristics	partially dispersive	dispersive	non-dispersive

- The potential vulnerabilities of the Turkey Point containment to hydrogen combustion are more pronounced than for both Surry and Zion (due to larger power-specific maximum hydrogen mass and maximum hydrogen molar concentrations in Turkey Point).

- The containment design pressure and failure pressure for Turkey Point are 0.5 MPa (59 psig) and 1.10 MPa (145 psig), respectively (about 0.1 MPa (14 psi) higher than at Surry and Zion). This indicates that Turkey Point has a more robust containment than either of the two reference plants.

- Late containment pressurization (as well as ex-vessel fission product releases) by noncondensable gases is influenced by the gaseous content of the basemat concrete. The concrete type at Turkey Point is limestone. At Surry, it is basalt, and, at Zion, it is limestone.

TABLE 2 Comparison of Containment Capacities

Containment	T. Point	Surry	Zion
Design Pressure	0.5 MPa (59 psig)	0.41 MPa (45 psig)	0.41 MPa (47 psig)
Failure Pressure	1.10 (145 psig)	0.97 MPa (126 psig)	1.02 MPa (134 psig)

3.1.2 Audit of Licensee's Sequence Binning, Containment Event Trees, and Accident Progression Analysis

3.1.2.1 Plant Damage State Definition

The Plant Damage States (PDSs) binning attributes include:

- Core Melt Timing
  - Early Core Melt, Rapid Inventory Loss (LLOCAs) (less than 2 hours): ECCS fails in injection.
  - Delayed Core Melt, Slow Inventory Loss (SLOCAs & Transients) (2 to 6 hours): ECCS fails in injection.
  - Late Core Melt (greater than 6 hours): ECCS fails in recirculation.

- RCS Pressure
  - High (Pressure > 2000 psig)
  - Intermediate (200 < Pressure < 2000 psig)
  - Low (Pressure < 200 psig)
- Containment Pressure Boundary Status
  - Isolated
  - Unisolated/Bypassed
- Containment Safeguard Status
  - Sprays Operational, w & w/o fans
  - Sprays Operate in Injection, but fail in Recirculation (at core damage), w & w/o fans
  - Sprays Failed, w & w/o fans
- Cavity Condition
  - Flooded (RWST injected)
  - Dry

The PDS binning process appears reasonable. However, the Interfacing Systems LOCA (ISLOCA) and SGTR sequences, which are referred to as Containment Bypass (CB) states, are not listed either separately or together as part of the dominant plant damage states (i.e., in Table 4.6-27 of Reference 1). The ISLOCA and the SGTR damage states appear to have a significant frequency, and should be listed as part of the dominant plant damage states in Table 4.6-27, page 4-171.

ISLOCAs and SGTRs should be identified, classified, and reported in the PDS summary tables. Furthermore, fission product release characteristics of these sequences are sufficiently different (from other early failure modes) that would require a separate quantification. (Source term calculations for these sequences appear in several studies, including NUREG-1150 and NUREG/CR-4629.) ISLOCAs and SGTRs often dominate the early risk of severe accidents. Therefore, both their frequencies and radiological release characteristics should be reported. (Generic Letter 88-20 lists sequences that should be reported, including "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 BWR-3 or PWR-4 release categories of WASH-1400." It appears that ISLOCAs and SGTRs fulfill these criteria.

Table 3 provides a simplified binning approach to the Turkey Point plant damage states. For comparison, the Turkey Point IPE plant damage state designators are also listed.

3.1.2.2 Containment Event Tree Analysis

Probabilistic quantification of severe accident progression is performed, using an event tree/fault tree approach. The CET structure is simple and consists of the eight top events/questions listed below. (Note sample CETs on pages 4-76 and 4-77 of the submittal.)

- RCS depressurization
- Coolant recovery in vessel
- No vessel failure
- Early containment failure
- Debris bed coolability ex-vessel
- Late containment failure
- Fission product release
- Containment failure mode.

Two nodes in the tree refer to events that occur before vessel failure.

Node DP addresses the issue of RCS depressurization either through operator action or through the effect of natural circulation-induced hot leg or surge line breaks. However, in the actual quantification of the trees, no credit is taken for operator action to depressurize the RCS.

Node REC addresses coolant recovery in-vessel. Coolant recovery is assumed to be possible, due either to RCS depressurization leading to the operation of Low Head Safety Injection (LHSI) pumps, or through the recovery of AC power, which causes the safety injection pumps to operate. For all plant damage states, a conditional probability of 0.743 has been assigned for success of coolant recovery in-vessel. This split fraction value has not been justified in the Turkey Point IPE submittal.

TABLE 3 Plant Damage States for Turkey Point

Initiator	ECCS Failure	Fans					
		Y			N		
		Sprays			Sprays		
		Y		N	Y		N
		FC	FC'	F	C	C'	
A	E	1.97E-6 (V-B) [0.44%]					
	L	1.26E-5 (VI-B) [2.8%]	5.14E-7 (VI-A) [0.1%]	2.66E-6 (VI-D) [0.59%]			
S	E			1.29E-6 (I-C) [0.28%]			2.13E-5 (I-H) [5.2%]
	L	4.95E-7 (II-C) [0.1%]				3.4E-4 (II-E) [76%]	
T	E	2.37E-5 (III-C) [5.27%]		5.42E-7 (III-D) [0.12%]			6.92E-7 (III-H) [0.15%]
	L						
CB1		6.24E-6 [1.40%]					
CB2		1.27E-5 [2.82%]* ?					

*Entire SGTR CDF placed in CB2.

Total Internal CDF = 4.49E-4, CDF listed in Table 3.1 of Reference 1 = 4.28E-4

See legend for table on the next page.

Legend for TABLE 3

A - Large LOCA
S - Small LOCA
T - Transients
E - ECCS Failure in Injection
L - ECCS Failure in Recirculation
F - Fans Operational
C - Sprays Operational
C'- Sprays Available in Injection
CB1 - Interfacing LOCA sequences
CB2 - Unisolated SGTR events

In each individual box of the table, the numerical values are from Table 4.6-27, page 4-17, and the nomenclature is defined in Table 4.3-2, page 4-34, and Table 4.3-3, page 4-35.

I:	S-LOCA, SGC, ECCS Fails in Inject.(P ≈ 1000 psi)
II:	S-LOCA, SGC, ECCS Fails in Recirc. (350<P<1000)
III:	Transients, NSGC, ECCS Fails in Injec.(P>2000)
IV:	SLOCA, ECCS Fails in Recirculation (P<200)
V:	L-LOCA, ECCS Fails in Injection (P<200)
VI:	L-LOCA, ECCS Fails in Recirculation (P<200)
SGTR:	No Heat Sink, No HHSI (P ≈ 1400 psi)

Node VB raises a question about the possibility of vessel failure. If coolant is recovered in-vessel, a 50-percent chance is assumed that core damage would be arrested, thus preventing vessel failure for all plant damage states. It should be noted that the possibility of arresting core damage upon coolant recovery is a complex phenomenon that depends on the extent of core damage and the actual time at which the coolant was injected into the vessel. Detailed analyses were conducted as a part of NUREG-1150. The analysts compared the extent of core damage at the time the in-vessel recovery occurred. The results are reported in Reference 10. In addition, a remote possibility exists that adding water to a degraded core can lead to early containment failure (i.e., in-vessel fuel coolant interactions, and increased hydrogen production). The event tree in the Turkey Point IPE does not consider this possibility.

Node CFE addresses early containment failure. The method of determination of the split fraction for early containment failure has been described in some detail in the IPE document, but, based on responses to questions asked during the site visit and the actual split fractions presented in the CET, it is believed that the method prescribed in the document is not rigorously followed. The uncertainty in the containment pressurization loads appears to have been neglected. In addition, it appears that the scaling from Surry to Turkey Point to obtain the containment loads was not based on accurate scaling parameters (see Section 3.1.1). Finally, the choice of Surry as a reference plant might lead to an underestimation of loads due to High Pressure Melt Ejection (HPME).

Node DC addresses the coolability of debris in the containment. The fault tree shown in the IPE document (See Figure 4.5-7) appears to be incomplete. In addition, the fault tree does not include the case of gravity pours into water (low pressure at vessel breach) with no ex-vessel steam explosion. NUREG-1150[7] experts assigned a low conditional probability of debris bed coolability to this case. On the other hand, in the Turkey Point IPE, a split fraction of 0.42 to 0.44 for coolability of molten debris was assumed. This estimate is not supported in the submittal, or by recent experimental data from Sandia National Laboratories. Furthermore, a molten pool/debris bed depth criterion for coolability on the concrete floor was also suggested in Appendix 1 to the IPE Generic Letter[3].

Node CFL addresses late containment failure. As in the case of Node CFE, the actual method of determination of the split fraction (based on discussions during the site visit) was not the same as that described in the IPE submittal.

One important concern about quantification of the CET is that a large fraction of the sequences that entail coolant recovery in-vessel with no vessel breach leads to late containment failure even with fans and sprays operating (i.e., PDS IIIC). This may

introduce an artificial vulnerability that may not be applicable to all severe accident conditions, and may cause problems if the present IPE submittal is used as a basis for the Turkey Point accident management process.

Node FPR addresses fission product removal. **Node CFM** addresses containment failure mode. The fault tree presented in Figure 4.5-10 of the submittal is valid only for large-sized failure (rupture).

A similar CET is depicted for impaired containment scenarios in Figure 4.5-2. However, this tree is not used to actually evaluate the accident sequences for the impaired containment. Instead, the original CET for the intact containment is used until the top event that raises questions about early containment failure is encountered. At this node, a conditional probability of 0.001 is assigned to containment isolation failure. The supporting analyses for assignment of this split fraction are neither documented nor discussed in the submittal. However, as part of the site visit, the review team was shown analyses that supported the 0.001 split fraction.

CET top events are developed using fault trees that represent the relationship of severe accident processes, systems operation, and operator actions. (See pages 4-78 through 4-89 of the submittal.) The logic trees are quantified by classifying basic events as follows:

- **Type 1:** Phenomena-related, subject to large uncertainties
- **Type 2:** Phenomena-related, influenced by plant-specific features
- **Type 3:** System-related, defined by PDSs or human response issues
- **Type 4:** True of false; dependent on previous CET event node.

In the quantification of Type 3 questions, it is desirable to take into account Equipment Qualification (EQ) issues relating to the impact of a degraded core/containment environment on successful operation of the Engineered Safety Systems (e.g., fans, sprays).

Table 4.6-29 of the submittal provides a description of each basic event. (See pages 4-173 through 4-182.) These event types appear to include relevant severe accident uncertainty issues. Section 4.6.8.3 provides the values of conditional probabilities for most of the basic events. The conditional probability estimates were assigned based on results obtained from MAAP calculations, and other PRAs. For the most part, NUREG-1150 results are used to assign the conditional probabilities. But several deviations are noted. For instance, in the Turkey Point IPE, the conditional probability that the hot leg and the surge

line remain intact, given a high-pressure accident sequence (Event PRHLSLOK in the fault tree for node DP) is 0.175 (unlikely event). This estimated split fraction is outside the NUREG-1150 expert-assigned range of 0.3 to 0.5.

3.1.2.3 Accident Progression Analysis

In the Turkey Point IPE, the dominant plant damage states are assessed deterministically using the MAAP computer code. The MAAP calculations include both "baseline" analyses and "sensitivities." The (key) MAAP input parameters are discussed in Section 4.6.4 of Reference 1. However, the technical bases for the selected parameters, for the most part, are either nonexistent or controversial. It would have been desirable, as part of the sensitivity calculations, to acknowledge the possibility of exceeding the considered range of parametric values. Sensitivity analyses have been performed by varying a few of the more significant parameters listed in Section 4.6.4 of Reference 1, as discussed below.

The MAAP parameters varied in the Turkey Point IPE include: (1) coefficient of cavity flooding (determines debris dispersal), (2) debris cooling CHF constant (determines heat transfer from debris to water), (3) fraction of debris that participates in DCH and (4) compartments participating in DCH. Several MAAP code parameters of interest have not undergone sensitivity analyses. The developers of MAAP code have recommended that sensitivity analysis of these parameters be included as a part of the IPE process[8]. The parameters might include the time required to fail the RPV lower head (after core plate failure), containment failure area, long-term revaporization rate from the primary system, and water penetration into regions packed with solid debris.

Sensitivities on the time required to fail the reactor pressure vessel lower head provide insights on the effect of debris-water interactions on RCS repressurization, and subsequent impact on containment. For accident sequences where the long-term revaporization from the primary system dominates fission product release, varying the core dump fraction provides insights on the impact of heating loads on the fission product revaporization, and RCS heatup. Allowing reflood and water ingress into the debris in-vessel will permit investigation of the hazards and benefits of recovery of a damaged core. Performing all the sensitivity analyses listed in Reference 8 of the submittal would have been helpful in assessing the uncertainties associated with severe accident progression as part of the CET analysis.

3.1.2.4 Fission Product Release Bins

The results of CET analyses lead to an extensive number of end-states, which are in turn binned for source-term analyses. This process is analogous to the one for defining PDSs for level 1 and level 2 analyses. Outcomes of the CETs are classified into a manageable number of releases, which are characterized by similarities in accident progression and source-term characteristics. The definition of release classes, categories and bins should contain as much information as possible on the accident sequence signatures and the status of the containment systems. However, the possible number of release bins to be evaluated increases almost exponentially with the degree of detail included in the bin definitions. As an example, the NUREG-1150 analysis of Surry includes information on 10 different parameters of interest in accident progression. This results in an extensive number of release bins.

The Turkey Point IPE submittal[1] defined 27 release bins, including intact containment/recovered states. (Note Table 4.7-1, pages 4-206 and 4-207 of the submittal.) The lack of separate bins for bypass sequences is a concern. (Bypass states were binned together with other early releases.)

For simplification, the 27 bins were condensed into a smaller number of bins, considered adequate for gaining some understanding of the resulting Containment Matrix (C-Matrix). (The submittal C-Matrix is displayed in Table 4.6-30, pages 4-185 and 4-186 of the submittal.) The key bin attributes include the following:

- Containment status and failure mode

  NCF: Intact containment

- Containment failure time. Two failure times are considered; early failure and late failure.

  A,B,C: Late containment failure
  D,E: Early containment failure

- Occurrence of CCI

  A: No CCI (no VB)
  B: No CCI
  C: CCI
  D: No CCI
  E: CCI

Table 4 lists the C-matrix for the dominant PDSs in the Turkey Point IPE. It can be seen from the C-matrix that the late failure is the dominant mode of containment failure for all PDSs. Additionally, a large fraction of the sequences that entail

TABLE 4 Turkey Point IPE Containment Matrix

TIME OF CONT FAILURE	LATE			EARLY		INTACT
		CCI?		CCI?		
	No VB*	No	Yes	No	Yes	
PDS**	A	B	C	D	E	NCF
SLC' (IIE)		0.09	0.51	<0.01	0.01	0.38
TEFC (IIIC)	0.21	0.35	0.42	<0.01	0.01	-
SE (IH)	-	0.47	0.52	<0.01	0.01	-
ALFC (VIB)	0.37	0.27	0.36	-	<0.01	-
ALF (VID)	0.37	0.27	0.36	-	<0.01	-
AEFC (VB)	0.37	0.27	0.36	-	<0.01	-
SEF (IC)	-	0.47	0.51	<0.01	0.01	-
TE (IIIH)	0.21	0.01	0.77	-	0.01	-
TEF (IIID)	0.21	0.35	0.42	<0.01	0.01	-
ALFC' (VIA)	0.37	0.27	0.36	-	<0.01	-
SLFC (IIC)	-	0.01	0.51	<0.01	0.02	0.46

* In-Vessel Recovery
** The parenthetical PDS

coolant recovery in-vessel with no vessel breach are found to lead to late containment failure even with fans and sprays operating (e.g., plant damage state III C). This is an important area of concern and could have an adverse impact on the use of the present IPE submittal for application to accident management.

3.1.2.5 Radionuclide Release Calculations

The Turkey Point IPE radiological release estimates are based on the simplified approach of NUREG/CR-4551[10] (NUREG-1150), and NUREG/CR-4881[11]. Taking this approach, the radiological releases are decomposed into several phases and quantified based on several base-case calculations for key release and transport attributes. However, in implementing this approach to assess the Turkey Point IPE, the following modifications were introduced, which appear to be incorrect. (Note submittal values and definition of terms on pages 4-210 and 4-211 of the submittal, respectively.)

- The in-vessel contribution to total source term was reduced by a factor, EFAERCOR(i), which is defined as "escape fraction for aerosol agglomeration uncertainties." The aerosol agglomeration effects were already credited in arriving at the FCOV(i) parameter. Therefore, by introduction of this factor, the effect of aerosol agglomeration was double-counted.

- The ex-vessel contribution to total source term was also reduced by a similar factor, EFAERCCI(i). Here again, the effects of aerosol agglomeration were already included in FCONC (FCONCE & FCONCL) parameters. This again is double-counting.

- An escape fraction, EFLEAK, was introduced to discriminate between containment failures due to leakage and gross ruptures. It is not clear how EFLEAK values were estimated in the Turkey Point IPE because these values were not reported in NUREG/CR-4881 or other NRC documents. In addition, the parameter FRDH appears to have been misinterpreted as "Fraction of core participating in DCH," while it is actually the fraction of the ejected core mass participating in DCH.

The release fractions for five representative radiological groups from various studies of reference plants are listed in Table 4.7-2 (see page 4-208 of the submittal). Pool DF values of 10, 3, and 1 are recommended in NUREG/CR-4881. The numbers cited in Table 4.7-2 are not consistent with these recommendations. Furthermore, the following comments apply to the Turkey Point release calculation input constants of Table 4.7-3 (see page 4-210):

- FCOR values for Te are set to zero based on MAAP Te_2 releases; most of the released Te is expected to be present as TeO_2. This fraction for in-vessel release should differ from I and Cs only because of its dependence on Zr oxidation. Otherwise, it should be in the same range as I and Cs. Because IPEs are expected to be used for accident management applications, important accident signatures must be captured.

- The FCCID values (FCCID is the fraction of initial inventory released from the melt during CCI for dry cavity cases) are set to 0.3 for Te and 0.05 for Sr. Both these estimates are low. More reasonable values for the inventory of material remaining after vessel failure would be 1.0 for Te and 0.3 for Sr. Furthermore, the FCCID values for Sr are physically inconsistent with FCOR values. (FCOR is the fraction of the initial core inventory released from the fuel prior to vessel failure.) Ex-vessel releases should be at least as high as in-vessel releases.

In addition, the quantified source terms do not include ISLOCA and SGTR sequences. These PDSs are expected to be significant (in terms of frequency and release magnitude), and should be treated separately. Furthermore, although small as fractions of initial inventory, but large in terms of actual quantity of material, radiological estimates for more refractory species may be important to equipment survivability during severe accidents.

3.1.3 COMPARISON OF RESULTS WITH OTHER STUDIES

Table 5 shows a comparison of the conditional probabilities of the various containment failure modes set out in the Turkey Point IPE submittal and the Surry NUREG-1150 study. In NUREG-1150, separate results were shown for internal initiators as well as for all initiators (internal, fire and seismic events). It should be noted that the Surry seismic results shown here are based on the EPRI characterization of seismic hazards. The separation of bypass and early containment failure is tentative and awaits a response from FP&L.

The core melt frequency from internal events is about a factor of two larger than for Surry (using the core damage value after "charging pump modification"). The largest containment-response difference is that the Surry conditional probability for late failure is about 6 percent; the corresponding value for Turkey Point is about 60 percent. The high conditional probability for late containment failure at Turkey Point appears to be partially an artifact of the IPE assumed demarcation time of 24 hours after accident initiation. The containment failure time is generally based on the referenced time of <u>core damage initiation</u>.

Therefore, better insights may have been gained if the same reference time point had been used for all plant damage states.

3.2 OUTSTANDING ISSUES

At this writing, formalized Round 2 questions on the Back-End IPE are being forwarded to FP&L. These questions and requests for information are provided below. It is anticipated that, for the most part, the FP&L responses will be a documentation and clarification of matters discussed during the site audit, but not contained in the utility submittal[1]. As discussed in Section 1 of this report, hydrogen combustion and ISLOCA issues fall into this category. However, there is one relatively unexplored issue that needs further consideration by FP&L, namely, containment performance improvements (CPIs). (Hydrogen combustion and its control are central to CPI, although the program and its intent go beyond the issue.) The comments and questions below include ones about CPI.

TABLE 5 Containment Failure as a Percentage of Total CDF: Comparison to Surry NUREG-1150 Results

Containment Failure	Surry/NUREG-1150		Turkey Point IPE
CDF (per reactor year)	Internal 4.1E-5	Fire 1.1E-5	Total* 4.5E-4
Early	1	2	1.6
Late	6	29	58.8
Late/No Vessel Breach	NA	NA	2.9
Bypass	12	0	4.4+
Isolation	***	***	0.1
Intact	81	69	32.2

* 1.0 X 10-4 (after modification) per year
*** Included in Early Failure
\+ Tentative, pending FP&L Response

3.2.1 Comments on CPI

There has been no attempt to address containment performance improvements, much less recommend any procedural or hardware changes that would mitigate the consequences of severe accidents. Specifically, the FP&L IPE submittal does not appear responsive to the issues raised in the CPI program, or to reflect the specific guidance provided in Generic Letter 88-20, Supplement

No. 3[6]. The CPI program for PWR dry containments addressed five potential containment performance improvements: depressurization, cavity flooding, hydrogen control, mitigation of ISLOCA, and containment venting[12, 13, 14]. Although it is recognized that, of all containment types, the PWR dry is the least vulnerable to severe accidents, there may be opportunities to further reduce risk, such as that posed by hydrogen. The Generic Letter Supplement specifically cites hydrogen combustion:

> **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.**

Because each containment is different and the Turkey Point containment may be more susceptible to significant hydrogen burning than some of the plants studied in the CPI program, a more aggressive look at potential hydrogen vulnerability seems appropriate. Yet, from submittal page 6.0-1, note that "...the level 2 or 'Back-End' analysis was not examined for 'vulnerabilities'."

The Florida Power & Light submittal does not appear to support a vigorous accident management program that might lead to development of new "Back-End" emergency operating procedures or other operator action guidelines. Florida Power & Light claims that they cannot address Back-End fixes because of the large uncertainties in the Back-End analysis. Yet there is no uncertainty assessment (qualitative or quantitative) in the IPE submittal. One of the reasons for seriously considering containment performance improvements is to circumvent large uncertainties that otherwise compromise statements about the low-risk profile of the plant.

3.2.2 Round 2 Questions and Requests for Information

1. The subject of local hydrogen pocketing and detonation was addressed in discussion of the Back-End Analysis during the IPE Evaluation Team plant visit on November 19-21, 1991. Members of the FP&L staff stated that their evaluation of the post-accident timing and containment internal configuration indicated that the potential for pocketing and detonation is small. Please document these discussions and the reasoning for your conclusion that the potential for pocketing and detonation is small and may be neglected in the Back-End Analysis. What hydrogen combustion-related sensitivity analyses were performed? Why were they not included in the IPE submittal? (Hydrogen and associated activities are key concerns of the Containment Performance Improvement program.) Drawings of the containment showing the Instrumental Tunnel, Reactor Cavity Compartment, Loop Compartment, Annular Compartment and Containment Upper Compartment should be provided with potential release points

and vent paths from the release compartments to adjacent compartments indicated. Estimates of the vent area from release compartments should also be provided.

2. In order to conclude that FP&L has satisfied the intent of the IPE program, please evaluate the Back-End results to identify containment vulnerabilities. Please discuss the evaluation, methods and finding, and list the 10 most frequent causes of containment failure (e.g., vulnerabilities). Consider these vulnerabilities in light of candidates for containment performance improvement identified in the CPI program documentation. If you believe that specific improvements are not warranted, please state why. If you think that some ideas have potential and you plan to integrate them into the upcoming accident management, please state so.

3. During the IPE review team site visit to Turkey Point clarifications were made by FP&L personnel about information provided in the formal submittal that helped the reviewers better understand the IPE. Please provide clarifications that address the following so that they might be included in the record:

   a. The EPRI document[8] (your Reference 4.6-2), "Recommended Sensitivity Analyses For An Individual Plant Examination using MAAP 3.OB," was used extensively in the submittal. Please provide a summary of what values were used from this report in your base-case MAAP analyses and your sensitivity analyses. For example, were all the values listed in Appendix A to this EPRI document used in your base cases?

   b. The method used to determine the timing or probability of containment failure was not clear in the submittal. Please provide a narrative that describes the process. Include the criteria that were used to determine if the containment failure was a rupture or a leak and the uses made of Figure 4.6-2 through Figure 4.6-5. There was confusion as to how the containment failure was calculated, how the determination was made and whether it was a rupture or a leak. The method used to determine the timing or probability of containment failure was not clear in the submittal. This includes confusion over the definition of "24-hour period." Why measure the success time at 24 hours from initiation of the accident, and not at the beginning of core damage? Are the following statements correct?

      - If MAAP calculates a failure at, say, 26 hours, then the pressure at 24 hours is compared to the failure probability curve and the probability plugged into the FT/CET. (No failures go beyond

24 hours. This, however, seems inconsistent with values on Page 4-168.) Early failures are all ruptures and the late ones are evenly divided between leaks and ruptures. Note that the MAAP hole size is constant.

c. Some of the containment logic trees are incomplete, that is, some basic events are described in the text narrative (e.g., pages 4.0-127, 128) but are not on the trees themselves (e.g., Figure 4.5-7). Please provide complete logic trees.

d. For the radionuclide release calculations described in Section 4.7, please summarize the origin of the values used in the calculations. Were they determined from specific MAAP runs for Turkey Point, from MAAP best-estimate input values, or from recommendations made by SAIC or found in NUREG-1150?

e. The analyses of the ISLOCA and unisolated SGTRs in the submittal were confusing. Please provide a narrative of how they were analyzed, what the PDS frequencies are estimated to be, and what the estimated fission product releases to the environment are.

f. It is confusing to have figures in the submittal that are not used. Please either state the purpose of Figure 4.5-2, or remove it. Also, please provide the correct text for Section 4.6.8.4.2.

4. In several areas of the Back-End analysis, values were used as input without justification. For many, justification is unnecessary, as these values are accepted by the technical community as a whole. However, there are values important to the outcome of the IPE that remain controversial. Of course you can choose the value you believe to be appropriate, but a justification is warranted. Please provide a justification for any values used that could be viewed as controversial. Include justifications for the following: (1) The basis for not releasing the Tellurium in-vessel (EPRI document, IPE Reference 4.6-2) is not correct. It is correct that the Te is held back by unreacted Zr. However, in a typical accident sequence, 30 to 50 percent of Zr is oxidized in-vessel. Therefore, similar quantities of Te are expected to be released in-vessel. It is not necessarily conservative to assume all Te is to be released ex-vessel. Why not use the NUREG/CR-4881 recommended values for in-vessel?, (2) Why were refractory species dropped from consideration in the radiological source-term assessment? (3) What is the justification for the values selected for FCCID (0.3 for Te and 0.05 for Sr)? (4) Please provide your rationale for source-term reduction factors EFAERCOR, and EFAERCCI.

5. What is the basis for assignment of split fraction to the node "coolant recovered in-vessel" (REC)? The same split fraction (a success probability value of 0.743) is assigned to all PDSs. Why?

   Explain why all the sequences that entail coolant recovery in-vessel and no vessel breach lead to late containment failure? (Generally, even for in-vessel coolant injection, one does not expect a late containment failure for damage states that include operability of containment heat removal systems.)

6. What is the approximate probability of flooding the reactor cavity, conditional on a core melt?

4. REFERENCES

1. "Turkey Point Plant Units 3 and 4 Probabilistic Risk Assessment - Individual Plant Examination," Final Report, prepared by Florida Power & Light, June 1991.

2. Responses to questions, and "Attachment for Questions," Florida Power & Light, (no date).

3. "Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR Part 50.54(f)," U.S. Nuclear Regulatory Commission, Generic Letter No. 88-20, November 23, 1988.

4. "Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities, 10 CFR Part 50.54(f)," U.S. Nuclear Regulatory Commission, Generic Letter No. 88-20, Supplement No. 1, August 29, 1989.

5. "Accident Management Strategies for Consideration in the Individual Plant Examination Process," U.S. Nuclear Regulatory Commission, Generic Letter 88-20, Supplement No. 2, April 4, 1990.

6. "Completion of Containment Performance Improvement Program and Forwarding of Insights for Use in the Individual Plant Examination for Severe Accident Vulnerabilities," U.S. Nuclear Regulatory Commission, Generic Letter No. 88-20, Supplement No. 3, July 6, 1990.

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

8. "Recommended Sensitivity Analyses for an Individual Plant Examination Using MAAP 3.0B," Gabor, Kenton & Associates, Inc., for Electric Power Research Institute, March 14, 1991.

9. "Status of Implementation Plan for Closure of Severe Accident Issues and Status of The Individual Plant Examination Program," U.S. Nuclear Regulatory Commission, SECY-91-084, March 28, 1991.

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

11. Nourbakhsh, H., M. Khatib-Rahbar, and R. E. Davis, "Fission Product Release Characteristics into Containments under Design Basis and Severe Accident Conditions," NUREG/CR-4881, March 1988.

12. Kelley, D. L., et al., "Quantitative Analysis of Potential Performance Improvements for The Dry PWR Containment," NUREG/CR-5575, Idaho National Engineering Laboratory, August 1990.

13. Gido, R. G., et al., "PWR Dry Containment Parametric Studies," NUREG/CR-5630, Sandia National Laboratories, April 1991.

14. Yang, J. W., et al., "Hydrogen Combustion, Control, and Value-Impact Analysis for PWR Dry Containments," NUREG/CR-5662, Brookhaven National Laboratory, June 1991.

APPENDIX

IPE EVALUATION AND DATA SUMMARY SHEET

4.1 Plant Data and Plant Description

4.1.1 Plant-Specific Analysis

Yes

4.1.2 Unique Vessel Features

None found

4.1.3 Most Likely Vessel Failure Mode

Lower vessel head penetration tube failure

4.1.4 Unique Containment Features

None found, except relatively high probability for cavity flooding

4.2 Plant Models and Methods for Physical Processes

4.2.1 Codes Exercised during The Analysis

MAAP 3.0B 16

4.2.2 Referenced Codes or Models

Models embodied in the NUREG-1150 SURSOR Code

4.2.3 References on Phenomenological Treatment

NUREG/CR-4551, NUREG/CR-4881, NUREG-1150, NUREG-0956
IDCOR Issue Papers

4.2.4 Phenomenology Considered

HPME/DCH
In-vessel and ex-vessel steam explosion
Fission product revaporization
Reflooding of degraded core
Core concrete interaction

Hydrogen combustion (although not addressed directly in the CETs)

4.3 Bins and Plant Damage States

4.3.1 Number of Plant Damage States

52

4.3.2 Binning Factors

Time of core melt
RCS pressure at vessel breach
Containment safeguards systems
Cavity condition (flooded, dry)
RCS retention on structures for aerosols (not included)

4.4 Containment Failure Characterization

4.4.1 Structural Calculations

Limited calculations and NUREG-1150 as reference (Note submittal page 4-72)

4.4.2 Ultimate Containment Failure Pressure

Wet not provided, Dry 145 psig

4.4.3 Additional Radionuclide Transport and Retention Structures

Auxiliary Building retention was not credited.

4.5 Containment Event Tree

4.5.1 Conditional Probability That The Containment Is Not Isolated

1.0E-3

4.5.2 Number of CETs

2 different CETs for each PDS

4.5.6 Qualitative or Quantitative Treatment of Uncertainties

Not treated, although sensitivity studies were performed

4.5.7 C-Matrix

Provided (See submittal pages 4-184-4-185)

4.6 Radionuclide Release Characterization

4.6.1 Method to Determine Source Terms

NUREG/CR-4551, NUREG/CR-4881, NUREG-1150, NUREG-0956
MAAP results

4.6.2 Code or Referenced Source Term Analysis

NUREG-1150 (NUREG/CR-4551)
NUREG/CR-4881

4.6.3 Number of Release Categories

27

4.7 Containment Performance Improvements

PWR

BE; Not Addressed

4.5.3 Number of Nodes in Smallest And Largest CET

9

4.5.4 List CET Top Events

PDS -	Plant Damage State
DP -	Depressurized
REC -	Coolant Recovered In Vessel
VF -	No Vessel Failure
CFE -	No Early Containment Failure
DL -	No Ex-Vessel Coolable Debris Bed Formed
CFL -	No Late Containment Failure
FPR -	Fission Product Removal
CFM -	Containment Failure Mode

4.5.5 Dominant Containment Failure Mode for SBO

Late failure by overpressurization

4.5 <u>Accident Progression and CET Quantification</u>

4.5.1 Calculated or Template Containment Loads

Template loads from Surry (NUREG-1150)
Additional loads based on MAAP

4.5.2 Technique Used to Treat Equipment Survivability

None (See assumptions made in FP&L response to BE-3, Reference [2].)

4.5.3 Equipment Identified as Susceptible to Severe Accident Environments

None identified (See assumptions made in FP&L response to BE-3, [2].)

4.5.4 Dominant Contributors to Containment Isolation Sequences

Not provided

4.5.5 Dominant Contributors to Containment Bypass Sequences

Interfacing LOCA and some SGTR sequences