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DUANE ARNOLD ENERGY CENTER TECHNICAL EVALUATION REPORT ON THE INDIVIDUAL PLANT EXAMINATION BACK-END SUBMITTAL

Hanry A. Wagage

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SCIENTECH, Inc.
11140 Rockville Pike
Rockville, Maryland 20852

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E. EXECUTIVE SUMMARY

SCIENTECH performed a review of the back-end portion of the Iowa Electric Light and Power Company's Individual Plant Examination (IPE) on the Duane Arnold Energy Center (DAEC).

E.1 Plant Characterization

DAEC consists of a General Electric Company designed BWR-4 with a Mark I containment. It has a designed thermal output rating of 1658 MW and a gross electric power generation limit of 589 MW. DAEC received its operating license on February 22, 1974, and began commercial operation on February 1, 1975. The DAEC containment has an ultimate failure pressure of 140 psig.

E.2 Licensee's IPE Process

The DAEC IPE consists of the following four elements to meet Generic Letter 88-20: 1) front-end analysis, 2) back-end analysis, 3) consideration of safety features and plant improvements, and 4) IPE utility team and internal review.

For the back-end analyses, the IPE team used containment event trees, fault trees, and an approach developed by the Nuclear Management Resource Council. The team evaluated the systems, phenomena, and operator actions pertinent to containment performance during severe accidents.

E.3 Back-End Analysis

For DAEC, the IPE team calculated a total core damage frequency (CDF) of 1.5 E-5 per year. The conditional probabilities of containment failure, given core damage, were 41 percent for early failures, 29 percent for late failures, and 0.6 for bypass. The probability of little or no release at DAEC was calculated to be 29 percent. The frequency of early, high releases was 1.3E-6 per year. These results are from Revision 3 of the Probabilistic Safety Analysis and appear in the utility responses to NRC requests for additional information. The "contributor to containment failure" results, described below, are from the original submittal [1] because the values calculated in Revision 3 are not reported in the utility responses. [2] The utility did not report any major changes to the back-end analysis in its responses [2] and therefore the results are assumed to be valid.

The main contributor to containment failure was overpressurization (49 percent of CDF) followed by containment vent (29 percent of CDF). The drywell liner melt-through and energetic failure modes contributed less than 1 percent to the CDF. The IPE team assumed drywell liner failure to be a certainty in the accident sequences where the cavity stayed dry.

E.4 Generic and Containment Performance Improvement Issues

DAEC took the following actions in response to the recommendations of the Containment Performance Improvement Program. In addition, the utility plans to install a hardened piped vent.

- Several alternate injection sources that can provide external water sources to the RPV or drywell sprays were included in the DAEC Emergency Operating Procedures. Alternate water injection was included in a number of accident sequences as a potential method of preventing core damage, preserving containment integrity, or flooding containment. If alternate water injection could be made perfect (failure probability of 0) for all sequences, the base CDF would be reduced by 1.8E-8 per year and the high/early release decreased by a factor of 5.

- The safety relief valves were found to be reliable for reactor pressure vessel depressurization. Alternate hardware modifications to further improve the reliability were found not to be justifiable.

- The BWROG Rev. 4 EPGs were incorporated into EOPs. Iowa Electric participated in the BWROG Emergency Procedures Committee development of Rev. 4 and has been involved since then in examining potential changes.

E.5 Vulnerabilities and Plant Improvements

The DAEC defined vulnerability using the following criteria:

- Are there any new or unusual means by which core damage or containment failure occurs as compared with those identified in other PRAs?

- Do the results suggest that the DAEC core damage frequency would not be able to meet the NRC's safety goal for core damage?

- Are there any single failures of components that lead directly to a core damage state (not including the common cause failure of multiple components of similar types)?

Based on the answers to the above, Iowa Electric did not find any vulnerability at DAEC.

The back-end insights gained through the conduct of this IPE, as well as suggestions for possible improvements/strategies, were:

- Termination (in accordance with procedures) of RPV injection when the containment pressure exceeds a set limit can lead to core damage and a subsequent containment challenge. The prudency of terminating water injection to the containment under any circumstance for which core degradation may be aggravated should be evaluated.

- As an ex-vessel recovery action, the use of containment sprays and drywell sprays in lieu of low-pressure coolant injection appears to be most useful in response to degraded core conditions. Prioritization of injection systems could be included in future accident management development.

- Initiation of drywell sprays before RPV breach would preclude debris attack and failure of the drywell shell for some of the accident sequences that would allow or call for spray initiation before vessel breach. (Class IIIC, Class V, and Class ID sequence types would not allow or call for spray initiation before vessel breach.) Consideration of changes to EOPs allowing the use of drywell spray initiation as well as removing any ambiguity regarding the diversion of injection sources away from the RPV when adequate core cooling is not assured (i.e., low reactor water level) could be included as part of future accident management development.

- Drywell sprays offer an additional alternative to controlling the drywell temperature to avoid premature containment failure. Relaxation of the restrictions on the use of the drywell sprays in the drywell spray initiation curve of the EOPs may be a possible future accident management item to develop.

- EPG directions with regard to containment flooding sequences can result in the highest consequences at the earliest time. Future accident management strategies should provide guidance to the operator on protecting containment and cooling debris using methods that do not require venting of the RPV and avoid using the drywell vent unless no other alternative exists.

E.6 Observations

SCIENTECH observed the following in the DAEC IPE back-end analysis:

- The IPE team appears to have responded adequately to the recommendations of the Containment Performance Improvement Program.

- Although not explicitly stated in the submittal, the DAEC IPE team appears to have documented the radionuclide release categories appropriately for accident sequences exceeding the Generic Letter 88-20 screening criteria.

- The DAEC appears not to have performed a thorough internal peer review of the back-end portion of the IPE.

- The DAEC IPE team appears to have taken significant credit for the back-end operator actions and the back-end results were driven by these operator actions. For example, the operator action to flood the containment resulted in a relatively low conditional probability (given core damage) of drywell shell failure of 1%. Drywell venting which was performed to release combustible gas from the containment had a relatively high conditional probability of 29%. A part of the sequences involving drywell venting resulted in early containment failure (a value not reported in the submittal).

SCIENTECH noted the following strengths in the DAEC IPE back-end analysis:

- The front-end back-end dependencies appear to be treated well.

- The DAEC IPE team incorporated several plant-specific features into the MAAP assessment methodology.

- The DAEC IPE team performed a comprehensive assessment of phenomenological uncertainties of severe accident progression.

- The descriptions of structures, systems, and components of the containment and reactor buildings are comprehensive.

- The Summary of Insights demonstrates a thorough knowledge of the impact of severe accidents on structures, systems, components, and operations.

SCIENTECH noted the following weakness in the DAEC IPE back-end analysis:

- It was difficult to audit how the CETs were quantified. Quantification specifics were not provided in the original submittal. However, the quantification process and values for an important top event were provided in the DAEC response to the staff's RAI. This indicates that the DAEC did employ an appropriate quantification methodology.

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 Duane Arnold Energy Center (DAEC) Individual Plant Examination (IPE) submittal. [1, 2] This TER was prepared to comply with the requirements for IPE back-end reviews of the U.S. Nuclear Regulatory Commission (NRC) in its contractor task orders, and adopts the NRC review objectives, which include the following:

- To help NRC staff determine if the IPE submittal provides the level of detail requested in the "Submittal Guidance Document," NUREG-1335

- To help NRC staff assess if the IPE submittal meets the intent of Generic Letter 88-20

- To complete the IPE Evaluation Data Summary Sheet

In July 1993 SCIENTECH delivered a draft TER for the back-end portion of the DAEC IPE submittal to the NRC. Based in part on this draft submittal, the NRC staff submitted a Request for Additional Information (RAI) to Iowa Electric Light and Power Company on January 6, 1995. Iowa Electric Light and Power Company responded to the RAI in a document dated June 26, 1995. [2] This final TER is based on the original submittal and the response to the RAI.

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

1.2 Plant Characterization

DAEC consists of a General Electric Company designed BWR-4 with a Mark I containment. DAEC has a designed thermal output rating of 1658 MW and a gross electric power generation limit of 589 MW. DAEC received its operating license on February 22, 1974, and began commercial operation on February 1, 1975.

The primary containment consists of the traditional inverted light bulb steel drywell and steel torus wetwell design with a suppression pool of water typical of the Mark I design. The containment was designed for 56 psig internal pressure, has a total free volume space of 306,400 ft^3 (144,000 ft^3 in the drywell including the vent system and 162,400 ft^3 above the water pool in the torus), and a minimum of 58,900 ft^3 of water in the suppression pool.

The secondary containment consists of four subsystems, which are the reactor building, the building isolation control system, the standby gas treatment system, and the offgas stack. The secondary containment system surrounds the primary containment system and is designed to provide secondary containment for the postulated loss of coolant accident.

The DAEC containment has an ultimate failure pressure of 140 psig.

2. CONTRACTOR REVIEW FINDINGS

2.1 Licensee IPE Process

2.1.1 Completeness and Methodology.

The DAEC IPE back-end submittal appears to be essentially complete with respect to the level of detail requested in NUREG-1335, and appears to meet the NRC sequence selection screening criteria described in Generic Letter 88-20.

The IPE methodology used is described clearly and its selection appears justified. The approach followed is consistent with Generic Letter GL 88-20, Appendix 1.

As noted in Section 2.3.1, page 2-3, for the back-end analyses, the DAEC IPE team used containment event trees (CETs), fault trees, and an approach developed by the Nuclear Management Resource Council (NUMARC). The team evaluated the systems, phenomena, and operator actions pertinent to containment performance during severe accidents.

For the front-end back-end interface and the back-end assessment, the IPE team relied on a set of general assumptions used in back-end PRA analyses. These are listed in Section 4.2.3.1 of the submittal, starting on page 4-64. In addition, Section 4.2.3.2 (pages 4-70 through -72) lists known, conservative assumptions used in the back-end analysis; Section 4.2.3.3 (pages 4-72 and -73) lists potential, nonconservative assumptions. One of the conservative assumptions was to use a containment failure curve, that was more limiting than the one calculated by Chicago Bridge and Iron (CB&I) on a plant-specific basis for DAEC and the one developed for Peach Bottom (capacities: used, 140 psig; DAEC plant-specific, 163 psig; and Peach Bottom, 148 psig). This decision could have resulted in slightly shorter times to containment failure than if the CB&I plant-specific DAEC curve had been used. In addition, there might have been sequences that would have resulted in "no containment failure" if the CB&I curve had been used to assign a containment ultimate capacity in this analysis. One of the nonconservative assumptions was to use a drywell equipment mass of 2.7 million pounds, which later appeared to the IPE team as an overestimate, resulting in additional heat sinks and longer times to reach high drywell temperature (i.e., which affects both containment failure and revaporization source term contributions). To correct this overestimate the IPE team performed sensitivity studies with a reduced equipment mass (1 million pounds) and factored the results into the radionuclide release results.

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

Being a single unit site, multi-unit considerations do not apply to DAEC.

The IPE team used walkdowns and an internal peer review to confirm the as-built, as-operated status of the DAEC. Introductory or general walkdowns were performed for areas outside the containment including the reactor building, the torus room, the turbine building, the pumphouse and intake structure, and the simulator. A human error analysis walkdown was performed, which covered the areas of the simulator and areas outside the control room in which operator actions were required.

Revision 3 of the PSA appears to reflect the plant design and operation as of December 1994 and the original submittal as of 1992 (RAI response, p. A-6)

2.1.3 Licensee Participation and Peer Review.

Based on information in Section 5 of the submittal, which describes utility participation and the internal review team, it appears that the DAEC did not perform an internal peer review of the back-end portion of the IPE. The in-house peer review team appears to have focused on the front-end portion only: "An independent in-house review committee was created to review the information contained in the initial preparation of the system notebooks." (Section 5.2.1, page 5-4)

Section 5.3.2, pages 5-7 through 5-46, lists the comments made during the final review and the responses by the DAEC to the internal review team and two review teams of the consultant companies who participated in the IPE (ERIN Engineering and Research; Gabor, Kenton and Associates). Of 83 comments listed, five were related to the back-end analysis.

2.2 Containment Analysis/Characterization

2.2.1 Front-end Back-end Dependencies.

As noted in Section 4.5.2, page 4-124:

> The DAEC IPE directly links the front-end to back-end portions of severe accident sequences through directly linked event trees. These trees convey the support state conditions throughout the front-end and back-end trees and include considerations of preventive or mitigative features, as well as timing considerations.

Section 4.3.3, pages 4-80 through 4-82 of the submittal, summarizes the specific aspects of the front-end back-end dependencies as follows:

- Equipment Failures in Front-End: The computer carried the information on failed equipment from the front-end to the back-end. This eliminated the need for back-end analysis of failed equipment, unless the equipment was going to be repaired or recovered. The equipment included support systems, accident prevention systems, and mitigation systems.

- Human Errors: A check was performed to ensure that recoveries following all Level 1 sequences that result from human error can be justified as consistent with operating staff recoveries.

- RPV Status: The CET analysis made use of the RPV pressure condition resulting from the front-end analysis.

- Containment Status: The CET made use of the containment status resulting from the front-end analysis, which included whether the containment failed, was intact, or was experiencing high-pressure conditions.

- Containment Isolation: The containment isolation was evaluated on a sequence-by-sequence basis using support system dependencies transferred from front-end analyses.

- Accident Sequence Timing: Differences in accident sequence timing were transferred with the front-end sequences.

- Thermal-Hydraulic Deterministic Assessments: The thermal-hydraulic code analysis represented variations in timing and assumptions about subtle sequence variations.

- Dual Usage: The analyses accounted for the dual usage of common water sources and common power sources in the front-end and back-end analyses.

- Mission Time: The analyses considered the mission time for the entire sequence, i.e., from initiating event to release point.

- Timing of Recovery: The analyses accounted for equipment and power recovery, ensuring no double counting.

In performing its analysis, the IPE team appears to have treated front-end back-end dependencies appropriately.

2.2.2 Sequences with Significant Probabilities.

Section 3.4.1, page 3-439, and Section 6.2, pages 6-2 through 6-8, describe sequences with a significant probability of occurrence. The DAEC IPE team identified five sequences that should be reported to the NRC according to the criteria in Generic Letter 88-20, Appendix 2. These five sequences had a combined CDF of 3.5E-6 per reactor year, or 44 percent of the total CDF. Table 4.6-3, page 4-198, gives a summary of the core damage accident sequence subclasses, including designators, definitions, and frequency of occurrence.

The DAEC IPE team appears to have evaluated appropriately those sequences with significant probability of occurrence.

2.2.3 Failure Modes and Timing.

Chicago Bridge and Iron evaluated the DAEC containment capacity under severe accident conditions. CB&I investigated the failures of the steel containment structure (drywell, drywell head, and torus), containment hatches, hatch seals, penetrations, and isolation valves. The actual containment failure curve used was more limiting, according to DAEC, than that calculated by CB&I (page 4-70). Section 4.4.4, pages 4-98 through 4-107, describes the following containment performance issues:

- Capacity at low temperature (below 500°F)
- Capacity at intermediate temperature (between 500°F and 800°F)
- Capacity at high temperature (above 900°F)
- Capacities for high-suppression pool temperatures, high containment pressures, and for high safety relief valve discharge rates.

Figure 4.4-3, page 4-104, shows the primary containment performance regime on a plot of pressure versus temperature. DAEC has a relatively small power rating relative to the containment size. As noted on page 4-269, the containment free-volume to core-power ratios are greater than those for the Peach Bottom facility.

The containment capacity was combined with the deterministic MAAP calculations to determine the timing and location of many of the containment failure modes. Figure 4.4-2, page 4-87, schematically shows how the MAAP deterministic results compared with the ultimate containment capacity.

The mean containment ultimate failure pressure was calculated to be 140 psig (Section 4.4.4.1, page 4-99).

The large DAEC containment penetrations have silicone rubber seals, which begin to fail at a temperature of 700° F in nonsteam environments. Under "dry" severe accident conditions where water is not available to cool the slumped core debris, containment heatup is expected. Under this scenario, the silicone rubber seals are expected to maintain resiliency up to 700° F; beyond 700° F the seals are assumed to fail. Under "wet" or steam severe accident conditions where debris cooling is available, the containment environment is expected to be saturated at the maximum containment pressure (~ 150 psig or ~ 500° F to 600° F) and the seals are assumed to fail between 500° F to 700° F. The outer seal would be somewhat protected from the environment even if the inner seal failed.

At 800° F, the flange seal material is expected to deteriorate significantly. At this temperature, the yield strength of the bolts also drops to the point that yield could occur around a pressure of 88 psig. This would increase the leakage area and could create an area similar to that of a rupture.

The DAEC containment failure modes are consistent with those identified in Table 2.2 of NUREG-1335.

2.2.4 Containment Isolation Failure

The DAEC IPE team modeled the containment isolation failure as the first node in the CET. The fault tree for containment modeled (Section 4.6.2.4, page 4-194):

> . . . containment hatches and large lines that penetrate the containment and open to the containment atmosphere (e.g., purge and vent lines). The fault tree considers automatic isolation signals, pre-existing open pathways, manual isolation, and component failures.

As indicated in Table 4.5-5, page 4-159, the containment isolation failure is defined as a line, hatch, or penetration opening with a diameter greater than 2 inches. The IPE modeled a containment isolation failure as a large (2 ft^2) failure in the drywell.

The IPE team calculated the DAEC containment isolation failure probability using a fault tree for containment isolation failure coupled with data for containment isolation failure as extrapolated by Pacific Northwest Laboratory for the NRC. [3] The frequency of isolation failure is not reported in the submittal. Table 4.8-6 of the submittal lists the following with regards to isolation failure:

> Containment isolation system is highly reliable. The operating experience of DAEC and other BWRs indicates that containment isolation is reliable and that early release due to containment isolation failure is a negligible contribution to risk.

The DAEC IPE team appears to have assessed and identified contributors to containment isolation failure.

2.2.5 System/Human Response.

Section 4.6.2.5, page 4-195 of the submittal, notes that the DAEC IPE team considered proceduralized operator actions (i.e., directed by current EOPs).

Table 4.6-2, page 4-196, lists the following proceduralized operator actions considered in the back-end analyses:

- Isolation of primary containment pathway, given failure of automatic isolation
- RPV depressurization
- Injection recovery
- Off-site power recovery
- Emergency diesel generator recovery
- Combustible gas venting
- Containment flooding
- Containment venting
- Manual alignment of alternate injection systems
- RPV venting.

The IPE team analyzed operator involvement in the following CET top events:

- Operator depressurizes RPV (IS)
- Core melt progression arrested in-vessel (RX)
- Combustible gas venting initiated (GV)
- Containment flooding initiated (FC)
- Drywell steel shell intact (SI)
- Venting initiated and successful (CV).

The DAEC IPE team appears to have taken significant credit for the back-end operator actions and the back-end results were driven by these operator actions. For example, the operator action to flood the containment resulted in a relatively low conditional probability (given core damage) of drywell shell failure of 1 percent. Drywell venting, which was performed to release combustible gas, had a relatively high conditional probability of 29 percent.

2.2.6 Radionuclide Release Characterization

Section 4.7, pages 4-224 through 4-272 of the submittal, describes radionuclide release characterization for the DAEC plant. As noted on page 4-224:

> The radionuclide release sequences determined from the CET evaluation that exceed the screening criteria frequency (i.e., reporting criteria) have been assessed to determine their radionuclide release magnitude.

The DAEC IPE team used the following to determine the radionuclide release magnitude (page 4-224):

- Existing Mark I radionuclide releases for a similar plant to characterize some release sequences

- Plant-specific DAEC calculations to both confirm the surrogate plant calculations and to fill in missing sequence calculations.

Based on discussion on another page in the submittal this surrogate plant appears to be Peach Bottom.

The DAEC IPE defined release categories based on release timing and release severity. The three release timing categories used were as follows:

• Early (E)	Less than 6 hours from accident initiation
• Intermediate (I)	Greater than or equal to 6 hours, but less than 24 hours
• Late (L)	Greater than or equal to 24 hours.

The five release severity categories used were as follows:

• High (H)	Early fatalities
• Moderate (M)	Near-term health effects
• Low (L)	Latent health effects
• Low-Low (LL)	Potential for latent health effects
• Negligible (No Release)	Less than or equal to the containment design base leakage.

The IPE team did not perform consequence analyses to determine the release fractions for the above categories. Instead the team reviewed the results of detailed consequence analyses from previous IDCOR studies, PRAs, and NRC studies and found the following CsI release fractions: High - greater than 10%, Moderate - 1 to 10%, Low - 0.1 to 1.0%, Low-Low - less than 0.1%, and Negligible - much less than 0.1%.

Table 4.6-5, page 4-202 of the submittal, lists the release frequencies for 13 accident classes under 13 release categories, defined as listed above. The DAEC used the MAAP computer code to calculate the source term.

As reported in the utility responses to NRC requests for additional information, the DAEC has a "large" release frequency of 1.3E-6 per year.

Although not explicitly stated in the submittal, the DAEC IPE team appears to have documented the radionuclide release categories appropriately for accident sequences exceeding the Generic Letter 88-20 screening criteria.

2.3 Accident Progression and Containment Performance Analysis

2.3.1 Severe Accident Progression.

The DAEC used the BWR Mark I version of the MAAP computer code (version 3.0B, revision 7.03) to calculate the containment survivability under the postulated severe accident conditions. Table 4.6-1, page 4-179, of the submittal lists a sample of four MAAP calculation cases. In Section 4.6.3.3, Figures 4.6-2 through -9, give the predicted drywell pressure and temperature transients. (These figures appear on pages 4-181 and 4-182, 4-185 through 4-188, and 4-190 and 4-191, respectively.)

Table 4.2-2, page 4-64 of the submittal, shows that the DAEC IPE team did not assess the following sequences or phenomena by using MAAP calculations, but did assess them using probabilistic analyses:

- Ex-vessel steam explosion
- Mark I shell failure
- Direct impingement-induced failure
- Direct containment heating
- Reactivity insertion during core melt progression.

In the sensitivity assessment for drywell shell failure, it was noted that such a failure, induced by molten debris, has the potential to be a beneficial effect, "creating a radionuclide release pathway through the reactor building with increased DF potential." (See page 4-385).

The DAEC IPE team appears to have used a reasonable process to understand and quantify severe accident progression and to have addressed the phenomenological uncertainties of accident progression.

2.3.2 Dominant Contributors Consistency with IPE Insights.

In Table 1, below, dominant contributors to DAEC containment failure are compared with those contributors identified during individual plant examinations performed at similar plants, and with the NUREG/CR-1150 PRA results obtained at Peach Bottom. No major differences exist among the various results.

Care should be taken in making such comparisons (as shown in Table 1 below) because the definitions of failure categories may vary from one IPE to another. Note, for example, the different definitions of "early" in Figure 4.7-5 on page 4-244 of the submittal.

Table 1. Containment Failure as a Percentage of CDF: DAEC Results Compared with other IPEs and with Peach Bottom NUREG-1150 PRA Results

Containment Failure	Fitzpatrick IPE	Oyster Creek IPE	Browns Ferry IPE	Nine Mile Point 1 IPE[1]	Peach Bottom/ NUREG-1150	Duane Arnold IPE[1] [2]
CDF (per year)	1.9E-6	3.2E-6	4.8E-5	5.4E-6	4.5E-6	1.5E-5
Early Failure	60	16	46	25	56	41
Bypass	na	7	na	0.5	na	0.6
Late Failure	26	26	26	62	16	29
Intact w/Vessel Breach	3	0	3	13	18	29[2]
Intact w/o Vessel Breach	11	51	25	na	10	na

[1]Containment failure time is defined relative to accident initiation.

[2]In the DAEC response to the NRC RAI Figure A-6 shows a conditional probability of containment intact of 30%, high/early releases of 9%, and all other releases of 61%. Because the probabilities of containment failure modes shown in this table were not explicitly given, they were computed from the release frequency values given in Table A-2 of the same reference. [2]

na Not available

2.3.3 Characterization of Containment Performance.

The DAEC IPE team categorized the containment challenges under accident conditions according to one of the following four regimes (Section 4.4.3, pages 4-94 and 4-97 of the submittal):

- Pressure-induced containment challenge
- Temperature-induced containment challenge
- Combined pressure- and temperature-induced containment challenge
- Dynamic loads.

The CETs are grouped into three categories: CET1—containment initially intact; CET2—containment initially failed or seriously challenged before core melt; and CET3—containment bypassed. Of the three CETs, CET1 had the most top events with a total number of 19 as follows (CET2 and CET3 had 14 and 5 top events, respectively):

- Core damage entry state, IA
- Containment isolated, IS
- Operator depressurized the RPV, OP
- Core melt arrested in-vessel, RX
- Combustible gas venting initiated, GV
- Containment intact, CZ
- Mark I shell failure precluded, SI
- Injection established to RPV or drywell, TD
- Containment flooding initiated, FC
- Containment intact during flood or RPV breach, CX
- Flood completed, FC
- Containment heat removal initiated, HR
- Venting imitated and successful, CV
- Suppression pool not bypassed, SP
- No large containment failure, NC
- Coolant inventory makeup, MU
- Drywell intact, DI
- Wetwell airspace breach, WW
- Release mitigated in reactor building, RB.

The quantification of the Level II IPE model merged all of the deterministic thermal-hydraulic calculations, the postulated containment failure modes, the assessment of the containment ultimate strength, the assessment of mitigation, and the probabilistic assessment of the likelihood of each.

The CET functional top event, "Containment Remains Intact (CZ)," was used to analyze the energetic phenomena leading to early containment failure. The IPE team concluded that successful prevention of early containment failure required the following (Table 4.5-5, p. 4-162):

- No direct containment heating (DCH was precluded if the RPV was depressurized)
- No in-vessel steam explosion (in-vessel steam explosions were precluded if either the RPV was at high pressure (> 100 psig) or the core did not fragment into fine particles before dropping on to the bottom head)
- No ex-vessel steam explosion
- No failure of vapor suppression (i.e., the suppression pool was not bypassed and vapor suppression success was guaranteed by having no more than 1 drywell to wetwell vacuum breaker failed open)
- No high pressure spikes sufficient to cause containment failure at the time of vessel melt-through (i.e., extreme pressure spikes were precluded if the RPV bottom head penetration failed locally; or the RPV remained at low pressure)
- No hydrogen deflagration or detonation (i.e., if the containment remained inert or effective combustible gas vent was operated successfully then hydrogen detonation or deflagrations was guaranteed not to occur)
- No RPV blowdown from high pressure with the suppression pool temperature above 240° F
- No recriticality due to an unusual core configuration that may be achieved during the melt progression.

If the above failure modes could not be prevented, containment failure was assumed to occur. The failure location was assumed to be in the drywell head region and was classified as a large failure. The probabilistic assessment used probability values based on industry or NRC studies. If a substantial energetic event occurred probabilistically, it was assumed to fail the containment.

The DAEC IPE team used the MAAP computer code to calculate the containment performance in CET sequence evaluation. Table 4.6-1, page 4-179, lists a sample of four MAAP calculation cases for four accident sequences representing four accident classes. Figures 4.6-2 through -9 give the predicted drywell pressure and temperature transients. The following is a summary of results:

Class IA: Loss of Makeup at High RPV Pressure with the Containment Initially Intact. The timing of key events for this sequence was as follows:

Core uncovered	0.69 hours
Initiation of core damage	1.06 hours
Initiation of core melt	1.28 hours
RPV failure/breach	3.38 hours.

In this sequence the containment remained intact and negligible releases occurred. A drywell pressure spike of 67 psig and a temperature spike of 580° F occurred at vessel breach, which did not cause containment failure. Following these spikes containment pressure and temperature increased slowly up to 20 hours into the accident and then started to level off and no containment challenge occurred.

Class ID: Loss of Adequate RPV Makeup at High RPV Pressure. The key events in this postulated sequence involved the plant response when the RPV had been successfully depressurized, but no injection was available to the RPV. The timing of key events for this sequence was as follows:

RPV depressurization	0.59 hours
Core uncovered	0.62 hours
Initiation of core damage	0.64 hours
Initiation of core melt	1.39 hours
RPV failure/breach,	1.91 hours
Containment failure	~27 hours
Radionuclide release	27 hours.

Containment failure occurred at the drywell, which was of large size. The radionuclide release magnitude was moderate. A pressure spike occurred at vessel breach, which was less than that for a high-pressure blowdown. Over the next 20 hours containment pressure and temperature rose until the primary containment failed 27 hours into the accident.

Class II: Loss of Adequate Containment Heat Removal. This accident involved core damage only after containment failure occurred. Containment heat removal was postulated to fail but the capability of injecting coolant into the RPV remained available until containment failure occurred. The timing of key events for this sequence was as follows:

Core uncovered	0.02 hours
Initiation of core melt	26.5 hours
RPV failure/breach	28.9 hours
Containment failure	25.0 hours
Radionuclide release	28.9 hours.

Containment failure occurred at the drywell which was of large size. The radionuclide release magnitude was High. The containment failed at relatively low containment temperature 25 hours after scram and loss of containment heat removal.

Class IV: ATWS Induced Containment Failure Followed by Core Damage. In this accident sequence containment failure was induced by a rapid increase in containment pressure, which preceded core damage. The timing of key events for this sequence was as follows:

Core uncovered	0.12 hours
Initiation of core damage	0.99 hours
Initiation of core melt	1.39 hours
RPV failure/breach,	4.20 hours
Containment failure	1.0 hours
Radionuclide release	1.0 hours.

Containment failure occurred at the drywell head, which was of large size. The radionuclide release magnitude was moderate. The containment failed "early," i.e., within 1 to 2 hours and core damage followed soon afterward.

2.3.4 Impact on Equipment Behavior.

Section 4.6.2.3, pages 4-192 through 4-194, describes equipment survivability in severe accident environments. Before taking credit for equipment operation under severe accident conditions, the IPE team assessed the capability of the equipment "to perform the function for a specific period of time considering exposure to temperature, pressure, aerosol loading, radiation, and moisture." The IPE team reviewed the results of research studies and equipment survivability tests of the following components:

- Cables
- Electrical penetration assemblies
- Electrical connections
- Solenoid valves
- Motor-operated valves
- Motor-driven pumps
- Motor control centers.

The DAEC IPE team appears to have considered the impact of severe accidents on equipment behavior.

2.3.5 Uncertainty and Sensitivity Analyses.

Section 4.2.3.1, page 4-67, of the submittal notes that the DAEC IPE team treated phenomenological uncertainties using MAAP sensitivity studies and insights from other studies. Section 4.9, pages 4-326 through -425, describes the DAEC sensitivity study of which the key results are summarized in Table 4.9-25, pages 4-417 through -425. As noted on page 4-337, the DAEC addressed sensitivities to the following:

- Core-melt progression
- In-vessel hydrogen generation
- RPV pressure at vessel failure

- Late CsI revaporization from the RPV
- Debris spread in containment
- Amount of debris retained in RPV
- Ex-vessel debris coolability
- Shell failure
- Containment failure location
- Containment failure area
- Reactor building effectiveness.

2.4 Reducing Probability of Core Damage or Fission Product Release

2.4.1 Definition of Vulnerability.

The DAEC IPE team defined vulnerability based on the answers to the following questions, which were referred to as "criteria" (Section 3.4.2, page 3-444):

- Are there any new or unusual means by which core damage or containment failure occur as compared with those identified in other PRAs?

- Do the results suggest that the DAEC core damage frequency would not be able to meet the NRC's safety goal for core damage?

- Are there any single failures of components that lead directly to a core damage state (not including the common cause failure of multiple components of similar types)?

Based on the answers, Iowa Electric found no vulnerabilities at DAEC.

2.4.2 Plant Improvements.

The back-end insights gained through the conduct of this IPE, as well as suggestions for possible improvements/strategies (Section 6.2.6, pages 6-9 and 6-10) were the following:

- Termination (in accordance with the procedures) of RPV injection when the containment pressure exceeds a set limit can lead to core damage and a subsequent containment challenge. The prudency of terminating water injection to the containment under any circumstances for which core degradation may be aggravated should be evaluated.

- As an ex-vessel recovery action, the use of containment sprays and drywell sprays in lieu of low-pressure coolant injection appears to be most useful in response to degraded core conditions. Prioritization of injection systems could be included in future accident management development.

- Initiation of drywell sprays before RPV breach would preclude debris attack and failure of the drywell shell for some of the accident sequences that would allow or call for spray initiation before vessel breach. (Class IIIC, Class V, and Class ID sequence types would not allow or call for spray initiation before vessel breach.) Consideration of changes to EOPs allowing the use of drywell spray initiation as well as removing any ambiguity regarding the diversion of injection sources away from the RPV when adequate core cooling is not assured (i.e., low reactor water level) could be included as part of future accident management development.

- Drywell sprays offer an additional alternative to control of the drywell temperature to avoid premature containment failure. Relaxation of the restrictions on the use of the drywell sprays in the drywell spray initiation curve of the EOPs may be a possible future accident management item to develop.

- EPG directions with regard to containment flooding sequences can result in the highest consequences at the earliest time. Future accident management strategies should provide guidance to the operator on protecting containment and cooling debris using methods that do not require venting of the RPV and that avoid using the drywell vent unless no other alternative exists.

2.5 Responses to CPI Program Recommendations

Generic Letter No. 88-20, Supplement No. 1, reiterates the following recommendations made during the Containment Performance Improvement Program (CPI) pertaining to the Mark I containments:

- Create alternate water supply for drywell spray/vessel injection
- Enhance reactor pressure vessel depressurization system reliability
- Implement emergency procedures and training.

Supplement No. 1 also notes that the above improvements should be considered in addition to improvements that stem from the evaluation and implementation of the hardened vent.

DAEC plans to install a hardened piped vent and the responses to other recommendations are as follows (pages Q#17-1 through -13 of the submittal [2]):

- Several alternate injection sources that can provide external water sources to the RPV or drywell sprays were included in the DAEC Emergency Operating Procedures. Alternate water injection was included in a number of accident sequences as a potential method of preventing core damage, preserving containment integrity, or flooding containment. If alternate water injection could be made perfect (failure probability of 0) for all sequences, the base CDF would be reduced by 1.8E-8 per year and the high/early release would decrease by a factor of 5.

- The safety relief valves were found to reliable for reactor pressure vessel depressurization. Alternate hardware modifications to further improve the reliability were not found to be justifiable.

- DAEC had incorporated the BWROG Rev. 4 EPGs into EOPs. Iowa Electric had participated in the BWROG Emergency Procedures Committee during development of Rev. 4 and has been involved since then in examining potential changes.

2.6 IPE Insights, Improvements, and Commitments

Section 4.8 of the submittal describes the IPE insights formulated based on specific DAEC calculations and probabilistic modeling of accident progression. The following is a summary as listed in Table 4.8-6:

- Containment isolation system is highly reliable. The operating experience of DAEC and other BWRs indicates that containment isolation is reliable and that early release due to containment isolation failure is a negligible contribution to risk.

- DAEC EOPs specify depressurization for most situations required. DAEC EOPs specify halting depressurization at 200 psig when turbine-driven systems are available but low-pressure injections systems are not.

- The DAEC EOPs are directed to the restoration of adequate core cooling even during degraded core states.

- Based on MAAP calculations, the containment or drywell spray may be used in lieu of low-pressure coolant injection in response to degraded core conditions.

- Drywell shell failure due to debris attack can be prevented if drywell sprays are initiated before reactor pressure vessel breach and the drywell floor is filled with water to quench the debris.

Iowa Electric plans to install a hardened pipe vent. No other commitments were made for plant or procedure improvement.

3. CONTRACTOR OBSERVATIONS AND CONCLUSIONS

As discussed in Section 2 of this TER, the DAEC IPE submittal contains a large amount of back-end information, which contributes to the resolution of severe accident vulnerability issues at the DAEC plant.

The key points of the SCIENTECH technical evaluation of the DAEC IPE back-end submittal are summarized as follows:

- In our opinion, the back-end portion of this IPE submittal provides a thorough, detailed, and well-written narrative on all aspects of severe accidents, containment (and reactor) building response, and anticipated radiological releases to the environment. The results are displayed in a variety of ways and in a graphical format that is easy to understand (see, for example, the series of figures on pages 4-218 through 4-223 and on pages 7-10 through 7-23).

- The quantitative results appear consistent with severe accident phenomenology and past analyses of similar BWRs using Mark I containments. However, it was difficult to track the quantitative analysis through (from PDS to release category) because CET top event split fraction values were not listed for the important sequences. The quantification specifics were not provided in the original submittal. However, the quantification process and values were provided for an important top event in the DAEC response to the staff's RAI. This indicates that the DAEC did employ an appropriate quantification methodology.

- The sensitivity analyses performed were extensive and informative (see pages 4-326 through 4-425), but they relied heavily on the MAAP code, with little indication of the validity of the code over the ranges of all the sensitivity parameters. The IPE used EPRI guidance for sensitivity studies.

- The IPE team appears to have responded adequately to the recommendations of the Containment Performance Improvement Program.

- Although it is not explicitly stated in the submittal, the DAEC IPE team appears to have documented the radionuclide release categories appropriately for accident sequences exceeding the Generic Letter 88-20 screening criteria.

- The DAEC IPE team appears to have taken significant credit for the back-end operator actions and the back-end results were driven by these operator actions. For example, the operator action to flood the containment resulted in a relatively low conditional probability (given core damage) of drywell shell failure of 1 percent. Drywell venting which was performed to release combustible gas from the containment had a relatively high conditional probability of 29%. A part of the sequences involving drywell venting resulted in early containment failure (a value not reported in the submittal).

4. REFERENCES

1. Iowa Electric Light and Power Company, "Duane Arnold Energy Center Individual Plant Examination Report," November 1992.

2. Iowa Electric Light and Power Company, "DAEC Response to Request for Additional Information on IPE," June 1995.

3. P. J. Pelto, K. R. Ames, and R. H. Gallucci, "Reliability Analysis of Containment Isolation Systems," Pacific Northwest Laboratory, NUREG/CR-3539, April 1984.

APPENDIX

IPE EVALUATION AND DATA SUMMARY SHEET

<u>**BWR Back-end Facts**</u>

Plant Name

Duane Arnold Energy Center

Containment Type

Mark I

Unique Containment Features

None found

Unique Vessel Features

None found

Number of Plant Damage States

13

Ultimate Containment Failure Pressure

140 psig

Additional Radionuclide Transport And Retention Structures

Suppression pool retention is credited. Extensive analyses of the reactor building retention capabilities were performed. However, as stated on page 4-70 of the submittal:

> Little credit is allowed for the reactor building DF. The . . . DF is limited to no more than a factor of 10 . . .

Note also the subsections, entitled "Reactor Building Effectiveness," beginning on page 4-152, and "Reactor Building Modeling Assumptions, beginning on page 4-385.

Conditional Probability That The Containment Is Not Isolated

Not available

APPENDIX
IPE EVALUATION AND DATA SUMMARY SHEET
(continued)

Important Insights, Including Unique Safety Features

- The EOPs that now dictate termination of the RPV injection when containment pressure rises above a set limit may need to be modified.
- Containment and drywell sprays may be used in lieu of low-pressure coolant injection into the RPV.
- Current Emergency Procedure Guidelines on containment flooding could result in high consequences.
- The likelihood that there will be a failure to depressurize is increased by the policy of immediately locking the automatic depressurization system, which puts the burden of depressurization on the operator.

Implemented Plant Improvements

None

APPENDIX
IPE EVALUATION AND DATA SUMMARY SHEET
(continued)

C-Matrix*

Accident Class	CDF (per year)	Early	Late	Intact
IA	4.14E-7	0.58		0.42
IB	1.92E-6	0.33	0.28	0.39
IC	1.49E-7	0.48		0.52
ID	4.80E-7	0.45	0.08	0.47
IE	1.01E-6	0.60		0.40
IIL	2.64E-7		1.00	
IIT	1.64E-6		1.00	
IIIB	1.24E-8	0.01		0.99
IIIC	2.62E-8	0.06	0.01	0.93
IIID	1.35E-7	1.00		
IVA	1.68E-6	1.00		
IVL	8.48E-8	1.00		
V	0			

* Notes:

1. For a comparable matrix of accident classes (PDSs versus Radionuclide Releases), see Table 4.6-5, page 4-202 of the submittal.

2. These results were obtained from the original submittal. [1] An updated matrix may be computed using the results provided on page Q#20-6 of the utility responses to NRC requests for additional information [2] which reports the Revision 3 results of the Probabilistic Safety Analysis.