SCIE-NRC-221-93

DRESDEN NUCLEAR POWER STATION
TECHNICAL EVALUATION REPORT
ON THE
INDIVIDUAL PLANT EXAMINATION
BACK-END ANALYSIS

Homayoon Dezfuli
James F. Meyer

Prepared for the U.S. Nuclear Regulatory Commission
Under Contract NRC-05-91-068-20
March 1995

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

9509010251 XA

TABLE OF CONTENTS

TABLE OF CONTENTS (cont.)

E. EXECUTIVE SUMMARY

E1. PLANT CHARACTERIZATION

Both of the Units at the Dresden Nuclear Power Station (Units 2 and 3) are BWR-3s with Mark-I containments. Each unit is rated at 773 MWE and is equipped with an isolation condenser.

E2. LICENSEE IPE PROCESS

The IPE conducted at Dresden was a PRA study based on the support state model. In performing a PRA, the total plant response to each severe accident scenario was modeled in plant response trees (PRTs) in order to integrate front-end and back-end analyses. The MAAP computer code was used to characterize success criteria, timing, and containment response.

The Dresden IPE back-end submittal reflects the plant as it was configured in January 1991. Although the "hardened" vent system was not yet operative at Dresden in 1991, the IPE team took credit for it. The hardened vent became operational for Unit 3 in 1993 and for Unit 2 in 1993.

The IPE team integrated accident management (AM) considerations into the IPE model. The team modeled the Emergency Operating Procedures (EOPs) using generic Boiling Water Reactor Owners Group (BWROG) symptom-based guidance. The IPE team also identified a class of core damage sequences referred to as Containment Success with Accident Management (CAM) sequences in which no containment failure occurs within 24 hours. After the first 24 hours, however, accident management is necessary to achieve long-term containment integrity.

E3. BACK END ANALYSIS

Important results reported in the Dresden IPE back-end submittal are as follows:

- The conditional probability of containment failure and/or containment venting with significant releases was 89 percent.
- The conditional probability of early containment failure with large releases was 3 percent.
- The core damage frequency (CDF) was 1.85E-5 per year.

- Failure of the suppression pool cooling (SPC) function accounted for 82 percent of the CDF.

The Dresden IPE team concluded that a single accident initiated by a loss of DC power, and in which the SPC failed, would constitute the dominant accident sequence, accounting for 44 percent of the CDF. A single accident initiated by a loss of DC power, and in which SPC failed due to operator error, would account for 9 percent of the CDF.

Modes of containment failure and their respective conditional probabilities at Dresden were reported to be the following:

- High-temperature structural failure after the containment was vented: 84 percent
- Rapid, high-pressure failure of the containment: 3 percent
- Late high-pressure and/or high-temperature failure: 2 percent

Based on these data, the conditional probability of the containment remaining intact is 11 percent.

E4. CONTAINMENT PERFORMANCE IMPROVEMENTS (CPI)

Based on Generic Letter No. 88-20, Supplement No. 1, the following CPI Program recommendations pertain to Mark-I containments:

- Create an alternate water supply for drywell spray/vessel injection (Procedure under consideration)
- Enhance reactor pressure vessel depressurization system reliability (No Action)
- Implement emergency procedures and training (Procedures under consideration in Accident Management Program)
- Evaluate and initiate operation of the hardened vent (hardened vent installed).

E5. VULNERABILITIES AND PLANT IMPROVEMENTS

The IPE team identified several enhancements that would reduce SPC failures. These are now under consideration for implementation. The major enhancement, which the NUMARC Severe

Accident Issue Closure Guidelines recommends, would be the introduction of a procedure to align low-pressure coolant injections (LPCI) or core spray (CS) pump suctions with the condensate storage tank (CST) when SPC cannot be established. This enhancement would ensure the retention of coolant injected into the reactor vessel when it would otherwise be lost. The coolant is lost when the net-positive suction head (NPSH) for the low-pressure emergency core cooling system (ECCS) pumps is not sufficient as the suppression pool water is heated. If this enhancement were implemented, the IPE team stated that the CDF would be reduced substantially. The team that was responsible for the Dresden Nuclear Station individual plant examination (IPE) back-end submittal concluded that no severe accident issues existed at Dresden that warranted immediate remedial action.

E6. OBSERVATIONS

In conducting the review, SCIENTECH noted the following from the Dresden Nuclear Power Station back-end submittal:

- Although the two units at Dresden could share mitigating capabilities, one unit was susceptible to a loss of DC power in the other unit.
- Given a core damage accident, the probability of venting the containment and not having a subsequent containment failure was very low.

Based on SCIENTECH's review, the following strengths and weaknesses in the Dresden Nuclear Power Station IPE back-end submittal are noted:

- The IPE back-end submittal is well structured and well written.
- As part of the IPE, the IPE team conducted several experiments to investigate lower vessel head cooling.
- The team appears to have used a systematic process to gain insights from the IPE.
- The IPE team did not fully describe the PRT structures in the back-end submittal.

- The IPE results indicate that the containment is sensitive to the loss of DC power. The team did not suggest in the submittal a preventive measure to reduce or eliminate this initiating event.
- No independent review of the IPE was performed.

I. INTRODUCTION

I.1 REVIEW PROCESS

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

- To determine if the IPE submittal provides the level of detail requested in the "Submittal Guidance Document," NUREG-1335
- To assess if the IPE submittal meets the intent of Generic Letter 88-20
- To complete the IPE Evaluation Data Summary Sheet

A draft TER for the Back-End portion of the Dresden IPE submittal was submitted by SCIENTECH to NRC on November 17, 1993. Based in part on this draft submittal, the NRC staff submitted a Request for Additional Information (RAI) to Commonwealth Edison on August 24, 1994. Commonwealth Edison responded to the RAI in a document dated October 28, 1994. This final TER is based on the original submittal and the response to the RAI.

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

I.2 PLANT CHARACTERIZATION

The Dresden containment is a BWR-3 Mark I design. The primary containment surrounds the reactor vessel and circulation cooling system. Any leakage from the primary containment will go directly to the reactor building. The primary containment consists of the following:

- Drywell. The free volume is 158,236 ft^3 with a gas space height of 102 feet. The drywell is a steel pressure vessel with a 66-foot-diameter spherical lower portion and a 37-foot-diameter cylindrical upper portion. The vessel is enclosed in reinforced concrete with a 2-inch gap between the steel shell and the concrete. The design pressure range of the drywell is 62 psig and -2 psig at 281 °F. The ambient temperature range is 135 to 150 °F.

- Pressure suppression pool torus (Wetwell). Used to control drywell pressurization under accident conditions.

- Interconnecting vent pipes. Eight circular vent pipes connect the drywell to the wetwell. The pipes enclosed in sleeves are provided with bellows to accommodate differential motion between the drywell and the wetwell. The pipes are connected to a vent header located in the air space of the wetwell.

The wetwell and drywell may be vented through use of a Standby Gas Treatment (SBGT) system or preferably through a 10-inch "hardened" vent to the 310-foot chimney.

In addition to the suppression pool, the IPE team used the following systems to model primary containment pressure control:

- Low-Pressure Coolant Injection (LPCI). This can be aligned with either the drywell or wetwell spray headers.

- Operator actions that are taken to restart the drywell coolers (Upon activation of the core spray system, the drywell cooler fans trip.)

- Drywell or wetwell venting.

II. TECHNICAL REVIEW

II.1 LICENSEE IPE PROCESS

II.1.1 Completeness and Methodology

The Dresden Station IPE back-end submittal is essentially complete with respect to the level of detail requested in NUREG-1335.

The Dresden IPE team used an integral approach to model the plant response from initiating event through the entire accident progression, including the containment response. This integrated approach combined the traditional Level I and Level II analyses into a single model. Each initiating event was tracked through a PRT to evaluate the success or failure of each plant system, operator action, and containment system. The PRTs developed for use in the Dresden IPE were complex and contained many top events. For example, the PRT for the general transient initiating event included 26 top events. To deal with complexity of the PRTs, the team broke them down into several subtrees. A complete list of the PRT top events appears in Table 4.1.3-4, pages 4-31 and 4-32 in Volume 1 of the submittal. The PRT structures are provided in Volume 2. Key support systems were modeled using support state methodology. In Section 4.5.3 of the submittal, the split fractions for support system event trees are provided. Also provided is the split fraction for PRT nodes for the dominant-100 core damage sequences.

The team used MAAP analysis to develop physical plant models to (1) define PRT nodal success criteria, (2) establish the timing of accident progression, and (3) determine accident outcomes. PRT outcomes (end states or plant damage states) were categorized as any of the following:

SCS -- Success

SAM -- Success with accident management, i.e., accident sequences that require accident management activities 24 hours after the core damage event in order to achieve an ultimately safe, stable containment state

CD -- Core damage

Sequences ending in core damage (i.e., CD endstates) each received a 5-character code to characterize their unique core and containment response characteristics. The team identified many CD plant damage states.

II.1.2 Multi-Unit Effects and As-Built As-Operated Status

The Dresden IPE back-end submittal reflects the plant as it was configured in January 1991. Although the "hardened" vent system was not yet operative at Dresden in 1991, the IPE team took credit for it. The hardened vent became operational for Unit 3 in 1993 and for Unit 2 in 1993.

Although the two units at Dresden could share mitigating capabilities, one unit was susceptible to a loss of DC power in the other unit.

II.1.3 Licensee Participation and Peer Review

Commonwealth Edison Company (CECo) engaged the Individual Plant Evaluation Partnership (IPEP) to support the conduct of the Dresden IPE. The IPEP participant companies were Westinghouse, Fauske and Associates, Inc., and TENERA. IPEP personnel performed the basic modeling and analysis. The CECo staff performed the success criteria analysis using MAAP and they conducted detailed reviews of the models, assumptions, and results. The sensitivity analyses that the team performed and reported on in the submittal were based on NUREG-1335 and EPRI-TR-100167 recommendations.

A "Tiger Team" composed of IPEP and CECo personnel evaluated the insights gained from the IPE. The IPE team stated that an IPEP Senior Management Support Team (SMST), whose members were not involved in the day-to-day conduct of the IPE, also reviewed the key insights and key results. On page 1-2 of the submittal, it is stated that:

> As noted in the initial CECo responses to the Nuclear Regulatory Commission (NRC) on Generic Letter 88-20, no separate independent review of the Dresden IPE was performed.

II.2 CONTAINMENT ANALYSIS/CHARACTERIZATION

II.2.1 Front-end Back-end Dependencies

The PRTs developed for the Dresden IPE reflect that reactor systems and containment systems were treated collectively. Using the PRT concept, front-end and back-end analyses were integrated into one model. The front-end back-end dependencies were accounted for in the structure of the PRTs.

The core damage frequency by plant damage state (PDS) is shown in Table 1.5.1-2, page 1-25 of the submittal.

II.2.2 Sequences with Significant Probabilities

The core damage frequency for Dresden Station was calculated to be 1.85E-5 per year. The IPE submittal states that, of this total, the frequency of core damage where the containment remained intact and was not vented was 2.1E-6 per year.

This implies that the frequency of core damage and containment failure (including venting) was

1.85E-5 - 2.1E-6 = 1.64E-5

Stated differently, the conditional containment failure, given a core damage accident, was approximately 89 percent.

Table 1.5.1-2 of the submittal lists the dominant core damage frequency accidents with their associated plant damage states. The top three dominant accident sequences are shown in Table 1 of this report.

Table 1

PDS	Description	Frequency (per year)	Contribution to CDF
DLCO	Loss of DC power with late core damage (6-24 hours) and SPC failure	1.06E-05	57.2
LLCO	Loss of offsite power (single or dual unit) with late core damage (6-24 hours) and SPC failure	3.27E-06	17.7
MLCO	Medium LOCA with late core damage (6-24 hours) and SPC failure	7.56E-07	4.1

The three sequences described in Table 1 accounted for approximately 80 percent of the total CDF. Each sequence described a plant damage state in which there was a low-pressure vessel melt-through with no water applied to the debris bed. The containment failed after having been vented previously. As can be seen from the table, the most dominant PDS involved the loss of DC power and subsequent loss of SPC.

Using the data contained in the submittal, which were based on the top-100 dominant core damage accidents, SCIENTECH attempted to construct the Dresden conditional containment failure probabilities. (These values, with minor changes, were later verified in the Commonwealth Edison 10/20/94 response to RAI.)

Table 2

Failure Mode	**Conditional probability of Containment failure**
Intact	11.2%[1]
Vented and failed late	84.2%
Late, high-temperature/high-pressure	1.5%
Early	3.0%
Bypass/isolation	0

II.2.3 Failure Modes and Timing

In the Dresden IPE, the fission product releases for core damage sequences represented on the PRTs were calculated using a time

---

[1]This value is for when the containment is intact and not vented for at least 24 hours. The value of 0.3% given on page 7-1, third paragraph, and in Table 7.1-3 on page 7-5 of the submittal is for when the containment is intact and not vented for 48 hours. (Note that the value of 0.3% was corrected to 0.4% in the 10/28/94 response to RAI.)

frame of 48 hours from the initiation of the accident. The IPE team extended the IPE window, which is 24 hours.

The Dresden containment fragility curve presented in the submittal shows that the mean failure pressure was approximately 105 psig for temperatures below 281 °F. Low-pressure containment failures were dominated by drywell head closure. If the containment were to fail at a relatively high pressure, the likely location would be the vent line bellows. On page 4-113, the IPE submittal provides mean failure pressures for critical locations as shown below:

Drywell shell	125 psig
Equipment hatch	>165 psig
Personnel airlock	150 psig
Mechanical penetrations	140 psig
Electrical penetrations	>150 psig
Drywell head closure	125 psig (leakage)
Vent line bellows	93-253 psig (leakage)
Wetwell shell	125 psig

Using the terms "unlikely" and "likely" in a conditional probability sense, the IPE team defined unlikely and likely failure modes in the IPE submittal, which are summarized below. Unlikely failure modes included steam explosion, vessel thrust forces, molten core-concrete attack, direct containment heating, thermal attack of containment penetrations, hydrogen combustion, and containment isolation. Likely failure modes were stated to be containment high pressure, containment high temperature, unscrubbed venting of the drywell, and liner melt-through. Table 4.3.3-1, page 4-112 of the submittal, summarizes the IPE team's phenomenological evaluations of all the containment failure modes postulated for Dresden.

- **Steam Explosion (unlikely)**. The IPE team concluded that the slumping of molten debris into the RPV lower plenum could not result in an in-vessel steam explosion. Moreover, the submittal states that an ex-vessel steam explosion would not threaten containment integrity.

- **Vessel Thrust Forces (unlikely)**. The submittal states that, based on a bounding analysis, this failure mode is highly unlikely. Even if vessel thrust forces did occur, they would not threaten containment integrity.

- **Molten Core-Concrete Attack (unlikely)**. The IPE team assessed the probability of a molten core-concrete attack within the Dresden containment, based on the most conservative scenario in which the debris is not coolable by direct contact with an overlying pool of water. The IPE team concluded that melt-through of the pedestal walls would not occur until well beyond the mission time of the IPE.

- **Direct Containment Heating (unlikely).** The submittal states that DCH is not a potential early containment failure mode. The ADS operation and "tightly packed" geometry of the drywell would inhibit the pressure rise associated with DCH.

- **Thermal Attack of Containment Penetration (unlikely).** In accident scenarios where debris coolability is absent, drywell integrity may be a concern. However, seal degradation generally will occur within a few days, if drywell airspace is not cooled.

- **Hydrogen Combustion (unlikely).** The team assessed the potential for H_2 combustion in a worst-case scenario involving an SBO event without initial containment inertion. The submittal states that H_2-combustion-induced containment failure can occur when AC power is recovered and the drywell sprays are initiated without first venting the wetwell. The team concluded that this situation can be avoided if wetwell venting is implemented before any attempt is made to use the drywell sprays.

- **Containment Bypass (unlikely).** According to the submittal, the frequency of interfacing systems LOCA was estimated to be on the order of 1E-10. For that reason, this failure mode was not analyzed.

- **Containment Isolation (unlikely).** Addressed in Section II.2.4 of this report.

- **Containment High Pressure and High Temperature (likely).** The submittal states that the potential for such failure exists in severe accidents where sufficient containment heat removal is not available. These accidents are primarily ATWS and station blackout events where venting is not possible, or is ineffective. The containment fragility curve in the submittal shows a probability of failure at 105 psig where temperatures are below 281 °F. The probability decreases to 62 psig at 500 °F. Because the containment gas temperature near the drywell head can reach the 300-500 °F range during severe accidents, leakage through the drywell closure can be expected at pressures less than 105 psig.

- **Liner Melt-through (likely).** The submittal states that liner melt-through will occur during a core-melt scenario in which the debris on the drywell floor is not cooled by an overlying pool of water. Once the debris is in contact with the shell, the shell will fail rather quickly. Because the sump in the pedestal region cannot hold all the possible molten debris, some of it may exit the pedestal region, make its way across the drywell floor, and eventually come into direct contact with the drywell steel shell. The IPE team treated this model of failure as part of its sensitivity analysis.

II.2.4 Containment Isolation Failure

The IPE team did not consider any containment isolation failures, although three conditions are described on page 4-119 of the submittal that may lead to containment isolation failures. The submittal states that none of these conditions would be likely because the Dresden containment is inerted during normal operation and would be isolated from the outset of a severe accident.

II.2.5 System/Human Response

The submittal states that the most significant operator-related contributions to the loss of heat removal from the containment are the failure to implement suppression pool cooling, the failure to make up to the shell side of the isolation condenser, and the failure to depressurize the reactor vessel when required. The second most likely sequence belonging to the most dominant PDS, i.e., DLCO (see Table 1 of this report) is related to a loss of DC power in one unit and the subsequent loss of suppression pool cooling due to operator error. This sequence accounts for about 9 percent of the total core damage frequency.

The IPE team identified a class of accident sequences referred to as Success with Accident Management (SAM) sequences, which do not result in core damage within 24 hours of the initiating event, but do require additional operator action after the first 24 hours in order to achieve a long-term safe, stable containment state. These sequences are primarily ATWS events. Table 4.6.5-4, page 4-252 of the submittal, lists the dominant operator failure modes for SAM accident sequences. The data show that these failures are the results of operator errors of omission, which, according to the submittal, can be remedied long after the errors are made.

II.2.6 Radionuclide Release Characterization

The IPE team performed source-term calculations based on the 100 highest-frequency sequences. To further reduce the fission product release calculations, these sequences were grouped into seven source-term bins.

For each bin, the dominant sequence belonging to the bin was selected for source-term evaluation. For example, the sequences listed in Table 1 of this report belong to a source-term bin labeled "CO." Table 4.5.5-3, page 4-191 of the submittal, tabulates the source-term analysis results for each source-term bin. The sequences were analyzed for 48 hours. The fission product releases were reported after 48 hours of elapsed sequence time. For the dominant source-term bin, i.e., CO, the following data were provided:

Fraction of clad reacted in vessel	0.1067
% Noble release	99.9
% Volatile FP release (CsI and RbI)	5.9
% Non-volatile FP release (SrO)	0.16
% Tellurium-based FP release (Te2 TeO2)	32.3

II.3 ACCIDENT PROGRESSION AND CONTAINMENT PERFORMANCE ANALYSIS

II.3.1 Severe Accident Progression

The IPE team used a special version of the MAAP code (MAAP BWR 3.0B Revision 7.03B) to develop its accident progression analysis. The submittal states that this version of the code was modified as to its in-vessel fission product retention capabilities. By varying the use of MAAP options, the IPE team investigated the influence of the core-melt progression model, and ex-vessel core debris coolability on containment failure timing.

Key results of the MAAP analysis for dominant sequences are summarized in Table 4.5.5-3 on page 4-191 of the submittal. For example, for the Sequence DLCO, (see Table 1 of this report) which is represented by the source-term bin, CO, the following data were provided:

Time of core uncovery (hr)	11.4
Time of core relocation (hr)	13.1
Time of vessel failure (hr)	16.2
Time of containment failure (hr)	27.5

Time of venting (hr)	16.5
Maximum drywell pressure (psig)	57.1
Maximum drywell temperature (°F)	668.

The IPE team performed sensitivity analyses, based on the following information sources:

- Table A.5 in NUREG-1335
- EPRI report EPRI-TR-100167, "Recommended Sensitivity Analyses for an Individual Plant Examination Using MAAP 3.0B"
- IPE analyst insights.

Tables 4.5.6-1 through 4.5.6-4 on pages 4-204 through 4-214 of the submittal tabulate the scope and results of the sensitivity analyses performed during the IPE.

II.3.2 Dominant Contributors: Consistency with IPE Insights

In Table 3 below, the conditional probabilities for the occurrence of the various containment failure modes are compared as they were set out in the Dresden IPE submittal, in several other BWR plants IPE submittals, and in the Peach Bottom NUREG-1150 study.

Table 3

Containment Failure	Fitzpatrick IPE	Oyster Creek IPE	Browns Ferry IPE	Duane Arnold IPE	Peach Bottom/ NUREG-1150	Cooper IPE	Dresden IPE
CDF (per year)	1.9E-6	3.2E-6	4.8E-5	7.8E-6	4.5E-6	7.1E-5	1.8E-5
Early failure	60	16	46	47	56	36	3
Bypass	na	7	na	0	na	0	0
Late failure	26	26	26	32	16	31	86
Intact	3	0	3	21	18	33	11
No vessel breach	11	51	25	na	10	na	na

Based on the submittal data, most of the core damage postulated for Dresden would occur late (6-24 hours). Rapid, high-pressure failure of the containment was calculated to make only a small contribution to the total CDFs. ATWS sequences would account for most of the rapid, high-pressure failure of the containment. Of the late containment failures, 84.2 percent would be associated with sequences in which the containment failed due to high temperature after having been vented previously. The contribution of sequences resulting in late, high-temperature/ high-pressure containment failures would be less than 2 percent. The IPE team concluded that the contribution of interfacing systems LOCA to CDF would be negligible. The most significant contributor to core damage frequency would be the loss of suppression pool cooling. The plant's capability to remove heat from the containment would be reduced severely if there were a loss of 125VDC in one unit that resulted in the loss of isolation condenser and suppression pool cooling.

II.3.3 Characterization of Containment Performance

The Dresden containment is a BWR-3 Mark I design. The primary containment surrounds the reactor vessel and circulation cooling system. Any leakage from the primary containment will go directly to the reactor building. The primary containment consists of the following:

- Drywell. The free volume is 158,236 ft^3 with a gas space height of 102 feet. The drywell is a steel pressure vessel with a 66-foot-diameter spherical lower portion and a 37-foot-diameter cylindrical upper portion. The vessel is enclosed in reinforced concrete with a 2-inch gap between the steel shell and the concrete. The design pressure range of the drywell is 62 psig and -2 psig at 281 °F. The ambient temperature range is 135 to 150 °F.

- Pressure suppression pool torus (Wetwell). Used to control drywell pressurization under accident conditions.

- Interconnecting vent pipes. Eight circular vent pipes connect the drywell to the wetwell. The pipes enclosed in sleeves are provided with bellows to accommodate differential motion between the drywell and the wetwell. The pipes are connected to a vent header located in the air space of the wetwell.

The wetwell and drywell may be vented through use of a Standby Gas Treatment (SBGT) system or preferably through a 10-inch "hardened" vent to the 310-foot chimney.

In addition to the suppression pool, the IPE team used the following systems to model primary containment pressure control:

- Low-Pressure Coolant Injection (LPCI). This can be aligned with either the drywell or wetwell spray headers.

- Operator actions that are taken to restart the drywell coolers (Upon activation of the core spray system, the drywell cooler fans trip.)

- Drywell or wetwell venting.

The nodes of the PRTs used to perform the IPE reflected both reactor systems and containment systems. In the review that is the subject of this report, SCIENTECH examined the "PRT for Loss of DC Power in Unit 2" to identify the type of nodes that the IPE team considered. Only the containment-related PRT nodes are listed below:

LP --	LPCI pumps
LV --	LPCI injection valves
OSPC --	Operator action to align for suppression pool cooling
SPC --	Hardware for suppression pool cooling
OCNTS --	Operator action to initiate containment spray
CNTS --	Hardware for containment spray
OVNT --	Operator action to vent containment
SVW --	Hardware for 2 IN wetwell vent
SVD --	Hardware for 2 IN drywell vent
LVW --	Hardware for 10 IN wetwell vent
LVD --	Hardware for 10 IN drywell vent
WW/DW --	Location of containment failure

The nodes shown above are typical of other PRTs.

II.3.4 Impact on Equipment Behavior

Equipment survivability is addressed in Section 4.4.5, page 4-157 of the submittal. Table 4.4.5-1 lists components located in areas that could become environmentally harsh following an accident. The submittal concludes that, based on survivability evaluations, these components would remain operative within the IPE window (i.e., 24 hours). The IPE submittal also contains a preliminary list of equipment that would be needed to carry out post-24 hour containment accident management following a core damage event. This equipment is listed in Tables 4.4.5-2 and 4.4.5-3, pages 4-160 and 4-161 of the submittal. The IPE team deferred survivability evaluations of this equipment to the future, as part of the implementation of an Accident Management Program.

The IPE submittal assumes that the status of equipment at 24 hours after a core damage event would apply throughout the 48-hour mission time.

II.4 REDUCING PROBABILITY OF CORE DAMAGE OR FISSION PRODUCT RELEASE

The team's quantitative assessment of the benefits of the proposed ECCS pump suction realignment procedural change is described in Section 6.2 of the submittal. Implementation of this change would reduce the core damage frequency by a factor of 5 to 3.7E-6 per year. The frequencies of the PDSs in which the containment were vented would be reduced by a factor of 8. The frequency of the PDSs in which the containment failed after having been vented would be reduced by a factor of 20. The IPE team concluded that this procedural change would have the most impact on the loss of DC power. Table 7.1-3 of the submittal compares the containment performance probabilities for the IPE-based model and for the model based on the ECCS pump suction alignment procedural enhancement.

II.4.1 Definition of Vulnerability

In Section 7, page 7-9 of the submittal, it is stated that:

> Although actions will be taken to comply with the NUMARC Closure Guidelines, it is concluded that there are no vulnerabilities for Dresden Station which require immediate attention to improve the plant risk profile.

The IPE team grouped all of the accident sequences with respect to the NUMARC Severe Accident Closure Guidelines. The grouping of the top-100 sequences is provided in Table 4.7.2-1, page 4-260 of the submittal. Only Class II accidents require action under the Closure Guidelines. The frequency of Class IB accidents calculated for Dresden fell just below the cutoff frequency for requiring implementation of Severe Accident Management Guidelines. The plant improvements that the IPE team proposed in response to the Closure Guidelines are discussed in the next section of this report.

II.4.2 Plant Improvements

Based on the IPE results and the NUMARC Severe Accident Issue Closure Guidelines, CECo is considering the implementation of two procedural enhancements of the Dresden Emergency Operating Procedures.

- **Enhancement in Dresden Emergency Operating Procedures (DEOP 100).** The purpose of the procedures is to maintain the reactor vessel level when the ability to remove heat from the suppression pool has been lost. The procedural change would enable the suctions for either LPCIs or the CS pumps to be aligned to the condensate storage tank when the NPSH limits for the ECCS pumps were reached. The reactor vessel level would then be maintained by intermittent operation of the ECCS pumps supplied from the CSTs.

- **Enhancement in procedures involving loss of all AC power.** The purpose of the current procedures is to maintain the operation of the isolation condenser during extended SBO sequences. The submittal states that this procedure could be enhanced by instructing operators to manually open the circuit breakers to the isolation condenser's 250VDC MOVs prior to depletion of the 125VDC batteries. This would allow for continued operation of the ICs, even under SBO conditions.

II.5 RESPONSES TO CPI PROGRAM RECOMMENDATIONS

Based on Generic Letter No. 88-20, Supplement No. 1, the following CPI Program recommendations pertain to Mark-I containments:

- Create an alternate water supply for drywell spray/vessel injection (Procedure under consideration)

- Enhance reactor pressure vessel depressurization system reliability (No Action)

- Implement emergency procedures and training (Procedures under consideration in Accident Management Program)

- Evaluate and initiate operation of the hardened vent (hardened vent installed).

With the exception of the recommendation regarding pressure vessel depressurization, all of the above are discussed in the back-end submittal as part of the IPE insights and AM program development.

II.6 IPE INSIGHTS, IMPROVEMENTS AND COMMITMENTS

In Section 5.3 of the submittal, the process is described that the IPE team used to gain insights into Dresden Station. Two key AM guidelines that CECo is now considering whether to implement are summarized below. Implementation of these AM guidelines would require hardware modifications to the plant.

- Prevention of reactor pressure vessel failure. Through its AM Program, the IPE team was able to verify that submersion of the bottom portion of the reactor could prevent the failure of the reactor vessel after relocation of the damaged core to the lower head, assuming that the RPV support skirt were modified to allow the egress of steam. This modification would require placing holes in the RPV support skirt near its junction with the RPV. The holes would provide for the escape of both air and steam from the skirt during pedestal flooding.

A related insight on the part of the team was that provisions could be made to flood the reactor pedestal independent of the rest of the containment. This isolated flooding, which would require structural modifications, offers several advantages. It would result in the rapid submergence of the RPV. It also would prevent the complete flooding of the torus, which is what the existing guidelines require. The suppression pool as a heat sink could be lost if the primary system ruptured following floodup.

- Alternate Sources of Containment Spray for Source-Term Reduction. In recognition of the potential for large fission product releases after SBO and ATWS sequences, the IPE team proposed an AM strategy to use an alternate source of containment spray to control the fission product releases. An alternate source of containment spray could be provided for an SBO in one unit via a cross-connection to the other unit's low-pressure core injection system or to the plant's fire protection system.

In Section 2.1.4.2 of this report, a procedural enhancement related to realignment of ECCS pump suction is discussed. In the commonwealth Edison 10/28/94 response to the staff of RAI, it is noted that of the 130 insights derived from the IPE, 82 are related to containment and containment systems. The enhancement above is number DR-057/IP, in response to RAI 34. On page 7-7, IPE states that:

Based on the NUMARC Severe Accident Closure Guidelines, action may be taken to implement a procedure to realign the suction of the ECCS pumps to the condensate storage tanks . . .

III. CONTRACTOR OBSERVATIONS AND CONCLUSIONS

The IPE team presented its methodology in a well-structured manner, appears to have searched for insights in a systematic way, and integrated accident management considerations into the IPE performance. The results of the IPE are well presented in summary tables. An independent, detailed review of the IPE might be worthwhile since the team used the support state/large event tree approach, which is inherently difficult to review at a high level.

In conducting the review, SCIENTECH noted the following from the Dresden Nuclear Power Station back-end submittal:

- Although the two units at Dresden could share mitigating capabilities, one unit was susceptible to a loss of DC power in the other unit.

- Given a core damage accident, the probability of venting the containment and not having a subsequent containment failure was very low.

Based on SCIENTECH's review, the following strengths and weaknesses in the Dresden Nuclear Power Station IPE back-end submittal are noted:

- The IPE back-end submittal is well structured and well written.

- As part of the IPE, the IPE team conducted several experiments to investigate lower vessel head cooling.

- The team appears to have used a systematic process to gain insights from the IPE.

- The IPE team did not fully describe the PRT structures in the back-end submittal.

- The IPE results indicate that the containment is sensitive to the loss of DC power. The team did not suggest in the submittal a preventive measure to reduce or eliminate this initiating event.

- No independent review of the IPE was performed.

IV. REFERENCES

1. "Dresden Nuclear Power Station Units 2 and 3, Individual Plant Examination Submittal Report," Volumes 1 and 2, Commonwealth Edison Company, January 1993.

2. Commonwealth Edison, "Dresden Nuclear Power Station Units 2 and 3 Response to NRC Request for Additional Information (RAI)," October 28, 1994.

APPENDIX

IPE EVALUATION AND DATA SUMMARY SHEET

BWR Back-end Facts

Plant Name

Dresden Station, Units 2 and 3

Containment Type

Mark I

Unique Containment Features

The suppression pool (wetwell) or drywell may be vented through either the Standby Gas Treatment (SBGT) system or directly to the 310-foot chimney through the 10-inch, hardened vent system. The latter system, which is the called Augmented Primary Containment Vent (APCV), was not installed at Dresden at the time of the IPE. However, for purposes of its examination, the IPE team assumed that the APCV was operational.

Low-Pressure Coolant Injections (LPCIs) can be lined up to discharge to either the drywell or wetwell spray headers for suppression pool cooling (under consideration).

Torus water and condensate storage tank inventory are available for transfer from one Dresden unit to the other (under consideration).

Dresden is equipped with an isolation condenser.

The capability exists for long-term ECCS injection via pump suction realignment to the CST (to be implemented via a procedural enhancement).

Unique Vessel Features

None were addressed.

Number of Plant Damage States

20 PDSs were defined for the top-100 core damage accident sequences.

Ultimate Containment Failure Pressure

105 psig

Additional Radionuclide Transport And Retention Structures

Secondary containment system (reactor building)

Conditional Probability That The Containment Is Not Isolated

Failure to isolate sequences was precluded from consideration during the IPE analysis because of the inerted condition of the Dresden containment.

Important Insights, Including Unique Safety Features

Although two units could share mitigative capabilities, one unit was susceptible to the loss of DC power in the other unit.

Given a core damage accident, the probability of venting the containment and not having a subsequent containment failure was very low.

The conditional probability of containment failure and/or containment venting with significant releases was 89 percent.

The conditional probability of early containment failure with large releases was 3 percent.

The core damage frequency (CDF) was 1.85E-5 per year.

Failure of the suppression pool cooling (SPC) function accounted for 82 percent of the CDF.

Implemented Plant Improvements

Of the several enhancements reported to be under consideration, none had yet been implemented, according to the submittal.

C-Matrix

Construction of a C-Matrix was not possible because supporting data were unavailable.

ENCLOSURE 4

DRESDEN UNITS 2 & 3 INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(HUMAN RELIABILITY ANALYSIS)