ERI/NRC 95-108

TECHNICAL EVALUATION REPORT OF THE WOLF CREEK INDIVIDUAL PLANT EXAMINATION BACK-END SUBMITTAL

FINAL REPORT

November 1995

Energy Research, Inc.
P.O. Box 2034
Rockville, Maryland 20847

Prepared for:

SCIENTECH, Inc.
Rockville, Maryland 20852

Under Contract NRC-04-91-068
With the United States Nuclear Regulatory Commission
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E. EXECUTIVE SUMMARY

This Technical Evaluation Report (TER) documents the findings from a review of the back-end portion of the Individual Plant Examination (IPE) of the Wolf Creek Generating Station.

E.1 Plant Characteristics

The Wolf Creek Generating Station (WCGS) is a four-loop, 3411 MW(t) Pressurized Water Reactor (PWR) unit of Westinghouse design housed in a large dry containment, and operated by Wolf Creek Nuclear Operating Corporation (WCNOC). The Wolf Creek plant is similar to the Zion PWR, except that the rated power is about 5% larger, and the containment free volume is about 12% smaller. However, the design of the cavity and instrument tunnel is similar to Surry. The design of the containment does not facilitate flooding of the reactor cavity.

E.2 Licensee's IPE Process

The methodology employed in the WCGS IPE submittal for the back-end evaluation is clearly described in the documentation. The IPE submittal is consistent with the level of detail discussed in the "Submittal Guidance Document", NUREG-1335. The front-end analyses used the fault-tree linking methodology, and the event trees were quantified using the GRAFTER computer code. Fourteen event trees, and a number of supporting fault trees, were developed. The calculated Core Damage Frequency (CDF), including both internal events and flooding, was 4.2 x 10^{-5} per reactor year. Systemic sequence screening criteria (i.e., all systemic sequences that have a CDF greater than 10^{-7} per reactor year, all bypass sequences that have a CDF greater than 10^{-8} per reactor year, and all sequences that contribute to more than 95% of the CDF and the total frequency of containment failure) were applied to front-end results that resulted in 54 core damage sequences from the front-end analyses. Each accident sequence was then linked directly to the Containment Safeguards Event Tree (CSET). The CSET is composed of three nodes that address the failure probability of fan coolers, containment sprays, and containment isolation. Thus, any accident sequence identified by the front-end analyses, could lead to six outcomes. A limited binning of accident sequences was performed, and the binning of the outcomes of the CSET leads to the quantification of the Plant Damage States (PDSs).

There are 6 possible outcomes for a CSET for each core damage sequence. However, after binning and applying a truncation frequency of 10^{-10} per reactor year, seventy-three non-zero PDS sequences were found to result. Thirteen of these sequences (using MAAP analyses) were found not to lead to core damage, and they were not considered further in the back-end analysis. In addition, two sequences involving containment isolation failure were also dropped, and a final of fifty-eight PDS sequences were found to remain.

Each PDS sequence was then linked directly to the Containment Event Tree (CET). The event tree is composed only of seven nodes, and in addition, these nodes are not developed using fault trees. The CET contains the following seven top events:

(1) Initiator (PDS),
(2) Containment isolation intact,
(3) Low pressure vessel failure,
(4) Late containment failure,
(5) Debris coolable,
(6) Containment heat removal, and
(7) Fission product scrubbing.

The same CET is quantified for all PDS sequences. As evident from the size of the CET, several phenomena are considered outside the event tree. All phenomena that can lead to early containment failure are considered outside the containment analyses, and are ruled out as not being a threat to the WCGS containment integrity. The most important of these phenomena are Direct Containment Heating (DCH), hydrogen combustion, and steam explosions. The licensee used calculations and results from the literature to argue that the conditional probability of containment failure from these energetic events was vanishingly small, and thus the licensee concluded that the events do not pose a threat to containment integrity.

With the exclusion of energetic events, the only challenge to the containment integrity is from containment overpressurization due to steam or non-condensible gas buildup (from core-concrete interactions). Basemat melt-through was considered, but it was concluded that within the 48 hour framework of the IPE, ablation of the concrete basemat was not possible.

The results of the CET analyses lead to an extensive number of end-states, which are classified into a manageable number of release categories, characterized by similarities in accident progression and source term characteristics. The main characteristics of the CET end-states considered when developing these release categories in the submittal were the time of containment failure, release mode (bypass or containment failure), and the fractions of the core inventory of volatile and non-volatile species. Eighteen possible release categories were developed; however, only seven of the release categories had frequencies greater than 10^{-8} per reactor year.

The MAAP 3.0B (version 17.02) code was the principal tool used to determine the source terms. The dominant contributors to the releases are the containment bypass sequences (Interfacing systems LOCA sequences and the steam generator tube rupture sequences).

E.3 Back-End Analysis

The submittal reports a CDF of 4.2×10^{-5} per reactor year. The leading contributors to core damage frequency are station blackout sequences (45% of the CDF), followed by loss of offsite power sequences (12% of the CDF), and two flooding sequences (16% of the CDF). Steam Generator Tube Rupture (SGTR) sequences and interfacing system LOCAs contribute to approximately 0.2% of the total CDF.

Results from the containment analyses show that, given core damage, the conditional probability of radiological releases to the environment (due to both containment failure and bypass) is 0.042. The IPE submittal has determined that the conditional probability of early containment failure is insignificant for the Wolf Creek plant. Late failure due to steam or non-condensible gas buildup and basemat melt-through contribute to 3.8% of the CDF. Containment isolation failure contributes to 0.1% of the CDF.

Table E.1 Containment Failure as a Fraction of Total CDF

Containment Failure Mode	Conditional Probability
Early Failure	0.001
Late Failure	0.038
Bypass (V)	0.002
Bypass (SGTR)	0.0003
Isolation Failure	0.001
Intact	0.958

The small probability of early containment failure calculated in the WCGS IPE submittal can be attributed to the licensee treatment of energetic events (DCH, steam explosions, hydrogen combustion). These phenomena have been considered and ruled out (based on plant-specific calculations or a literature review) as possible threats to containment integrity. The conditional probability of late containment failure is also smaller than other IPE submittals. The small probability of late containment failure is attributed to the following two reasons. First, the containment has a large cavity floor area, and therefore the submittal assumes that there is a high conditional probability of coolability of debris on the cavity floor by an overlying pool of water. Secondly, the concrete type in the WCGS plant is a basaltic aggregate. The generation of non-condensible gases were found to be very small for this type of concrete, and hence the conditional probability of late overpressure failure is calculated to be low.

The submittal makes use of the MAAP code to calculate the radiological releases. The releases are dominated by steam generator tube rupture sequences and an interfacing systems LOCA sequence. The releases calculated for these sequences are large, and it can be said that the risk profile for the WCGS plant is dominated by the releases from the containment bypass sequences.

E.4 Generic Issues and Containment Performance Improvement (CPI) Issues

One of the recommendations of the CPI program pertaining to PWRs with large dry containments was that the utility should evaluate the IPE results for containment and equipment vulnerabilities to hydrogen combustion (local and global), and point out any need for procedural and/or hardware improvements. The submittal documentation does not explicitly discuss the recommendations of the CPI program. However, in response to the NRC questions, the licensee discussed the recommendations of the CPI program, and their treatment of the CPI program recommendations, which was summarized by a position evaluation summary report.

The licensee performed a walkdown of the containment, and could not identify any likely locations for hydrogen pocketing. The licensee identified several openings between the lower compartment, steam generator compartment, and the pressurizer cubicle, to the upper compartment. Hydrogen deflagration in the WCGS containment was studied, and was ruled out and has been ruled out as a contributor to early containment failure. A simplified analysis was performed to evaluate the potential for transition from Deflagration to Detonation (DDT). The upper containment was modelled as an unconfined geometry. The lower and annular compartments were modelled as channels. The analysis concluded that flame acceleration and transition to detonation was unlikely in all three compartments. The impact of hydrogen combustion upon the functioning of fan coolers was analyzed, but ruled out on the basis that the resulting temperature transient in the upper compartment can last only a few minutes. Assuming natural convection heat transfer to the fan coolers, the temperature transient from a burn has to last more than ten minutes in order to heatup the fan coolers to the point of damage.

E.5 Vulnerabilities and Plant Improvements

The submittal does not define "vulnerability", particularly as related to containment analyses. However, the licensee concluded that there are no back-end vulnerabilities at WCGS. This conclusion is based on the following results of the IPE:

- The peak containment pressure due to DCH and hydrogen combustion is well below the tail-end of the containment fragility curve. In addition, the licensee argues that the phenomena such as steam explosions, direct attack of containment penetrations, and vessel rocketing, cannot pose a threat to the containment integrity.

- A containment walkdown was performed to identify any potential locations of hydrogen accumulation, and no such locations were identified. The licensee concluded that hydrogen deflagration and detonation are not possible containment failure mechanisms at the WCGS.

- A number of accident sequences lead to significant ablation of concrete basemat, and a small conditional probability of containment failure due to overpressurization. However, the conditional probability ($\sim$0.04) is so low that MCCI cannot be considered as a vulnerability.

- The bypass sequences are small contributors to the CDF ($\sim$0.2%).

WCNOC has identified modifications or improvements in seven areas, namely, high temperature qualified RCP seal rings, replacement of positive displacement charging pump by a centrifugal charging pump, provision of a switch to restore main feedwater (if auxiliary feedwater fails), study of equipment dependence on room cooling, replacement of emergency procedures associated with loss of component cooling water or service water, identification of procedural or hardware modifications to reduce the CDF due to internal flooding, and implementation of accident management guidelines. The only "improvement" relevant to the containment analyses

are the planned implementation of accident management guidelines. The Westinghouse issued the generic guidelines in 1994. The licensee is committed to an assessment of the SAM capabilities and to implement any enhancements by September, 1997.

E.6 Observations

The important points of the technical evaluation of the WCGS IPE back-end analysis are summarized below:

- The back-end portion of the IPE supplies sufficient information regarding the subject areas identified in Generic Letter 88-20.

- The WCGS IPE provides an evaluation of all phenomena of importance to severe accident progression in accordance with Appendix I of the Generic Letter. However, all phenomena that can lead to early containment f[illegible]e ruled out as not being a threat to WCGS containment integrity.

- Direct sequence-by-sequence linking of front-end accident sequences to the CET analyses was the methodology used for the back-end analyses. However, only sixty front-end accident sequences were considered for back-end analyses, and twelve MAAP simulations were performed for deterministic analyses. The extent of analyses is very limited.

- The depth of treatment of phenomenological uncertainty, and severe accident progression issues, is fairly sparse. The CET consists of only seven nodes, and these nodes are not developed any further. In addition, quantification of these nodes, consists mostly of assigning values of zero or one, based on the results of the front-end analyses.

- Several issues, such as AC power recovery, depressurization after core damage, in-vessel core coolability, thermally-induced failure of hot leg and steam generator tubes, operator actions after core damage, recovery of containment cooling systems, etc., have not been considered in the containment analyses. Some of these issues have been considered using sensitivity analyses.

- Steam generator tube rupture sequences have been identified by other IPEs to be the dominant contributor to releases in PWRs with a large dry containment. In the Wolf Creek IPE front-end analyses, SGTR sequences were determined to contribute to 1.5 % of the CDF. The licensee re-analyzed the SGTR sequences using the MAAP code, and determined that four of the five analyzed SGTR sequences did not lead to core damage within the 24 hour mission time. Hence, a majority of the SGTR sequences were excluded from the back-end analyses. Exclusion of the risk-dominant SGTR sequences leads to a reduction in radionuclide release frequency, a reduction in the magnitude of the radiological releases, and thus is a weakness of the IPE submittal.

Licensee personnel were involved in the back-end analyses, which were performed with only a limited help from outside contractors. The licensee has considered the failure of containment isolation system and containment bypass. Failure of electrical and mechanical penetrations at elevated temperatures were considered and ruled out. The licensee has addressed the recommendations of the CPI program, requested as part of the GL 88-20, Supplements 1 and 2. No vulnerabilities have been identified.

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NOMENCLATURE

ATWS	Anticipated Transient Without Scram
CDF	Core Damage Frequency
CET	Containment Event Tree
CSET	Containment Safeguards Event Tree
CHR	Containment Heat Rejection
CPI	Containment Performance Improvement
DCH	Direct Containment Heating
ECCS	Emergency Core Cooling Systems
EOP	Emergency Operating Procedure
EPA	Electrical Penetration Assembly
EPRI	Electric Power Research Institute
EVSE	Ex-Vessel Steam Explosion
GE	General Electric
GL	Generic Letter
IPE	Individual Plant Examination
ISLOCA	Interfacing Systems Loss of Coolant Accident
IVSE	In-Vessel Steam Explosion
LOCA	Loss of Coolant Accident
LPCI	Low Pressure Coolant Injection
MAAP	Modular Accident Analysis Program
MCCI	Molten Core Concrete Interactions
MSIV	Main Steam Isolation Valve
NRC	Nuclear Regulatory Commission
PDS	Plant Damage State
PRA	Probabilistic Risk Assessment
RC	Release Category
RCS	Reactor Coolant System
RHR	Residual Heat Rejection
RPV	Reactor Pressure Vessel
RWST	Reactor Water Storage Tank
SGTR	Steam Generator Tube Rupture
SRV	Safety Relief Valve
TER	Technical Evaluation Report
WCGS	Wolf Creek Generating Station
WCNOC	Wolf Creek Nuclear Operating Company

1. INTRODUCTION

This Technical Evaluation Report (TER) documents the results of a review of the Wolf Creek Generating Station (WCGS) Individual Plant Examination (IPE) Back-End submittal [1]. This TER 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 essentially provides the level of detail requested in the "Submittal Guidance Document", NUREG-1335,
- To assess if the IPE submittal meets the intent of the Generic Letter 88-20, and
- To complete the IPE Evaluation Data Summary Sheet.

This TER complies with the requirements of the contractor task order for review. The remainder of Section 1 of this report describes the technical evaluation process employed in this review, and presents a summary of the important characteristics of the WCGS nuclear power plant related to containment behavior and post-core-damage severe accident progression, as derived from the IPE. Section 2 summarizes the review technical findings, and briefly describes the submittal scope as it pertains to the work requirements. Each portion of Section 2 corresponds to a specific work requirement as outlined in the NRC contractor task order. A summary of the overall IPE evaluation, identification of IPE submittal strengths and weaknesses, and review conclusions are summarized in Section 3. Section 4 contains a list of cited references. Appendix A to this report contains the IPE evaluation summary sheets.

1.1 Review Process

The technical review process for back-end analysis consists of a complete examination of Sections 1, 2, and 4 to 7 of the IPE submittal. In this examination, key findings are noted; inputs, methods, and results are reviewed, and any issues or concerns pertaining to the submittal were identified. The primary intent of the review is to ascertain whether or not, and to what extent, the back-end IPE submittal satisfies the major intent of Generic Letter (GL) 88-20 [2] and achieves the four IPE sub-objectives. A list of questions and requests for additional information were developed to help resolve issues and concerns noted in the examination process, and were forwarded to the licensee. The licensee responses [7] were reviewed. The final TER is based on the information contained in the IPE submittal [1] and the licensee responses to the RAIs [7].

1.2 Containment Analysis

The Wolf Creek Generating Station (WCGS) is a single unit, four-loop Westinghouse Pressurized Water Reactor (PWR), jointly owned by Kansas City Power & Light Company, Kansas Electric Power Cooperative Inc., and Kansas Gas and Electric Company, and operated

by Wolf Creek Nuclear Operating Corporation (WCNOC). The rated thermal power of the plant is 3,411 MW.

A brief description of the WCGS containment and plant data are provided in Section 4.1 of the submittal. Figures 2.4-1 through 2.4-8 (of the submittal) illustrate some of the design features of the cavity and the containment that are important for severe accident progression. The WCGS containment building is a pre-stresses, post-tensioned concrete cylindrical structure with a hemispherical dome and a flat base. The concrete base is a 10 ft thick reinforced concrete slab, 154 ft in diameter, and founded 11 feet below the plant grade. The containment is lined with a welded carbon steel plate with a thickness of 0.25 inches.

The following plant-specific features are important for accident progression in the WCGS plant:

- The cavity is connected to the lower compartment of the containment through an instrument tunnel. The cavity floor area is 648 ft^2. The lower compartment configuration is similar to Surry. It has at least two floors between the instrument tunnel exit and the containment region that can trap debris dispersed from the vessel due to High Pressure Melt Ejection (HPME) after vessel breach. The two floors, seal table room floor, and the operating deck floor, are a major obstruction to debris transport to the upper compartment.

- The WCGS containment does not facilitate flooding of the reactor cavity. Even though water can drain from the upper compartment to the lower and annular compartment floors, the annular compartment floor is located 16 inches below the lower compartment floor, and a 6 inch curb surrounds the cavity manway. Thus, the annular compartment has to be flooded to a depth of 22 inches, and the lower compartment has to be flooded to a depth of 6 inches before water can enter the cavity. A substantial portion of the RWST inventory (~114,500 gallons) has to be injected before a limited flooding of the cavity region begins.

Table 1 Summary of Key Plant and Containment Design Features for the WCGS Plant

Feature	WCGS	Zion	Surry
Power Level, MW(t)	3,411	3,236	2,441
Free Volume of Containment, m^3	70,792	81,000	46,440
Containment Volume/Power, m^3/MW(t)	20.8	25	19
Failure Pressure, psig	127.6	134	126
Concrete Aggregate	Basaltic	Limestone	Basaltic

A brief comparison of the available plant and containment data between the Wolf Creek plant, and the Zion and Surry plants, are shown in Table 1. Section 4.1 of the submittal provides a brief discussion of these plant-specific features.

2. CONTRACTOR REVIEW FINDINGS

The present review compared the WCGS IPE submittal to the intent of the Generic Letter (GL) 88-20, according to the guidance provided in NUREG-1335. The responses of the licensee to the NRC team RAIs were also reviewed. The findings of the present review are reported in this section, and follow the structure of Task Order Subtask 1.

2.1 Licensee's IPE Process

2.1.1 Completeness and Methodology

The IPE submittal is consistent with the level of detail discussed in the "Submittal Guidance Document", NUREG-1335.

The methodology employed in the WCGS IPE submittal for the back-end evaluation is clearly described, and the IPE is logical and consistent with GL 88-20. The front-end analysis concludes with the integration of system fault trees, and the core damage event tree was quantified to obtain the core damage frequencies for a number of accident sequences. Fifty-four dominant accident sequences that correspond to more than 99.8% of the CDF were identified for further, back-end analysis. Each of these accident sequences was linked to a Containment Safeguards Event Tree (CSET) that includes three separate nodes to address the failure probability of fan coolers, containment sprays, and containment isolation. The binning of the outcomes of the CSET leads to the quantification of the Plant Damage States (PDSs). Probabilistic quantification of severe accident progression involved the development of a small Containment Event Tree (CET). The results of the CET analyses lead to an extensive number of end-states which are binned into release categories. The MAAP code is used to simulate the containment response and to quantify the source terms.

2.1.2 As-Built/As-Operated Status

The freeze date of the IPE analysis for the containment and plant systems is stated to be the end of calendar year 1991. Insofar as the containment systems are concerned, it appears that all the WCGS containment-specific features are modelled.

2.1.3 Licensee Participation and Peer Review of IPE

The IPE back-end analyses were performed by the Risk Assessment Group of WCNOC, which is a part of the Nuclear Analysis Division of the company. Other individuals from other divisions such as Operations, Training, Engineering, Safety, and Licensing, were responsible for developing and providing input to the risk assessment analysts. It appears that the licensee obtained limited assistance from Westinghouse and Bechtel, Inc. for back-end analyses.

An "independent" review team from the Nuclear Safety Engineering group performed a review of the IPE. In addition, Westinghouse personnel and the Union Electric IPE team performed

a review of the WCGS submittal. The review team identified concerns with the flooding analysis, ISLOCA analysis, and station blackout analysis. No comments on the back-end analyses were apparently provided by the review team.

2.2 Containment Analysis

This section provides a review of PDS binning, CET analyses, release category definitions, severe accident analyses, and the containment structural analyses in the submittal.

2.2.1 Front-End/Back-End Dependencies

The front-end analyses were performed using a fault tree linking procedure using the Westinghouse-developed WLINK computer code. A cut-off frequency of 10^{-15} per reactor year (and up to 5000 cutsets) was used for screening the accident sequences. Fifty-four sequences with frequencies greater than 10^{-8} per reactor year, and contributing to 99.8% of the calculated CDF of 4.2×10^{-5} per reactor year, were screened for use in containment analyses. Each accident sequence was then linked to a Containment Safeguard Event Tree (CSET). The CSET is composed of three nodes that address the failure probability of fan coolers, containment sprays, and containment isolation. Thus, any accident sequence identified by the front-end analyses, could lead to six outcomes. To avoid an excessive number of PDSs, a limited binning of accident sequences were performed, and the binning of the outcomes of the CSET leads to the quantification of the Plant Damage States (PDSs).

PDSs are a combination of the sequence identifier defined in the front-end analyses, and a letter that represents the availability of containment safeguards, defined as follows:

A No fan coolers and no sprays,

B Fan Coolers available, but no sprays,

C Sprays available, but fan coolers are not available,

D Coolers and sprays available, and

E Isolation failure.

There are 6 possible outcomes for a CSET for each core damage sequence. However, after binning and applying a truncation frequency of 10^{-10} per reactor year, seventy-three non-zero PDS sequences were found to result. Thirteen of these sequences (using MAAP analyses) were found not to lead to core damage, and they were not considered further in the back-end analysis. In addition, two sequences involving containment isolation failure were also dropped, and a final of fifty-eight PDS sequences were found to remain. The frequency of each PDS sequence, and the representative accident sequence analyzed using the MAAP code is listed in Table 4.3-1 of the submittal.

The top contributor to core damage frequency is PDS sequence SB08A, a station blackout sequence which contributes to 14% of the CDF. A control room flood sequence is the next dominant contributor to CDF (10.65%). A number of station blackout sequences and loss of offsite power sequences rank next in importance.

The PDS definition and binning are very simplified in the submittal, and provide limited information of interest to back-end analyses. The definition of PDSs does not convey important information such as availability or recovery of ECCS, system pressure at core damage, etc. In addition, it is noted that 16 of the top 73 PDS sequences are long-term and short-term station blackout sequences. These sequences could have been binned together easily without loss of generality. In the absence of such binning, it is difficult to review the results, since the containment response for each of these sequences have to be understood independently.

2.2.2 Containment Event Tree Development

Probabilistic quantification of severe accident progression is performed using an event tree methodology. However, the event tree is composed only of seven nodes, and in addition, these nodes are not developed using fault trees. Quantification of the events that comprise the CET, and the CET quantification results is discussed in Section 4.2 of the submittal. The CET is concise and contains the following seven top events:

(1) Initiator (PDS),
(2) Containment isolation intact,
(3) Low pressure vessel failure,
(4) Late containment failure,
(5) Debris coolable,
(6) Containment heat removal, and
(7) Fission product scrubbing.

The same CET is quantified for all PDS sequences. The first event node represents the entry state to the CET.

Containment Isolation Intact: This top event addresses the possibility of containment isolation failure in the WCGS plant. Since isolation failure is treated in the CSET, the PDS definition includes information on success or failure of containment isolation.

Low Pressure Vessel Failure: The purpose of this node is to quantify RCS pressure at vessel breach. The boundary between high pressure and low pressure is defined to be 400 psig. High pressure at vessel breach can lead to High Pressure Melt Ejection (HPME) and Direct Containment Heating (DCH). In addition, HPME can also increase the airborne fission product concentration, which can increase the radionuclide source term releases.

Late Containment Failure: This event node is used to identify the time of containment failure. Success of this event node is determined from the results of the MAAP calculations and

phenomenological evaluation summaries for those phenomena that are not modelled in the MAAP code.

A number of phenomena that can lead to early containment failure including hydrogen combustion, DCH, steam explosions, thermal attack of concrete penetrations, and thrust force on the vessel at vessel breach, have all been ruled out in the WCGS IPE submittal using phenomenological evaluation summaries. Hydrogen combustion was determined to be not a threat to lead to early containment failure, based on the MAAP results of in-vessel hydrogen generation, and the resulting containment pressure loads.

The licensee utilized a five-step procedure to determine the containment pressure loads due to DCH in a high pressure accident sequence [7].

1. Determination of the bounding primary system and containment conditions at the time of vessel breach,

2. Determination of the extent of debris entrainment that would occur in the cavity/instrument tunnel,

3. Assessment of the fraction of entrained debris which would escape into the containment after the directional change at the seal table,

4. Determination of the effects of possible hydrogen combustion, and

5. Calculation of the containment pressurization resulting from the postulated DCH event, including the impact of hydrogen combustion.

In response to the NRC RAIs [7], the licensee provided a qualitative description of the methodology used to determine the containment loads for a postulated DCH event. The results for their bounding calculation indicate a containment pressurization of 65 psig (5.5 bars absolute) with continuous hydrogen combustion, and 35 psig (3.5 bars absolute) without continuous combustion. Details of the calculations, including the estimates of the mass of debris in the lower plenum, fraction of the metallic constituent, and fraction of core debris dispersed to the containment, etc. are not provided in the submittal document [1] or the licensee responses to the NRC RAIs [7]. However, it should be noted that the NRC has recently completed a study of the DCH issue for Zion and Surry plants [8,9], and attempted to extrapolate the results to all PWRs of Westinghouse design [10]. The results show that the calculated maximum containment pressure load for Wolf Creek is about 72 psig (6 bars absolute).

Then, the licensee compared the containment pressure loads to the containment fragility curve. The fragility curve does not include points below 80 psig (where the conditional probability of containment failure is less than 10^{-3}), and hence, the licensee concluded that the probability of containment failure due to DCH is vanishingly small.

Steam explosions were considered in another phenomenological evaluation summary. The licensee concluded after a review of results from the literature that in-vessel steam explosions are not a credible threat to containment integrity. The cavity in the WCGS plant is excavated and buried into the ground, and thus, shock waves produced by fuel-coolant interactions in the cavity are not expected to be a threat to containment integrity. It should be borne in mind that the Wolf Creek containment does not facilitate flooding of the cavity.

Two other modes of containment failure, namely, direct attack of containment penetrations, and thrust forces on the vessel (leading to "vessel rocketing"), were investigated, and found to be implausible in the WCGS plant. In summary, the licensee has considered and ruled out all modes of early containment failure in the WCGS IPE submittal.

Debris Coolable: After vessel breach, the debris that exits the vessel accumulates on the cavity floor. If the entire core inventory is postulated to be uniformly spread on the cavity floor, the thickness of the debris layer will be less than 25 cm. Hence, the licensee concludes that if an adequate supply of water is available to ensure that debris is immersed, then debris coolability can be assumed. It should be noted that there is considerable uncertainty in the possibility of core debris coolability by a pool of overlying water, and on the debris spread on the cavity floor. The lack of treatment of uncertainties in debris coolability and in melt spread, and the assignment of a split fraction of 0 or 1 (depending upon the availability of water) without considering these uncertainties, is one of the shortcomings of the submittal.

Containment Heat Removal: It is assumed that the operation of one fan cooler, or one RHR pump plus heat exchanger is sufficient to remove the heat generation in the late phase of the accident. Based on the availability of the containment heat removal systems (as indicated by the PDSs), split fractions of 1 or 0 are assigned to this node. Failure of containment heat removal assures containment failure by overpressure in the late phase of the accident.

Fission Product Removal: Operation of containment sprays and the presence of a pool of water overlying the debris can result in fission product scrubbing. A review of PDS definition is sufficient to indicate the availability of sprays and the possibility of injection of RWST water.

The CET analyses in the WCGS is very abbreviated, and does not consider several issues that have been considered in other IPE submittals, such as recovery of AC power, in-vessel core coolability, induced failure of hot legs and steam generator tubes, operator actions, etc. However, it should be pointed that the licensee has attempted to treat some of the above phenomena using sensitivity analyses. The licensee has attempted to treat all phenomena of interest to severe accident progression, either in the CETs, or through the phenomenological evaluations. However, the detail of treatment of phenomenological issues of interest to severe accident progression in a PWR with a large dry containment, and uncertainties in these phenomenological issues, in the WCGS IPE submittal is very sparse.

2.2.3 Containment Failure Modes and Timing

The WCGS IPE submittal makes use of plant-specific calculations to determine the ultimate pressure capacity of the containment. The results of the structural analyses indicate that the total median failure pressure is 127.6 psig, with a lower bound (5%) and an upper bound (95%) capacities are 99 psig and 136 psig, respectively. At containment pressures below 123 psig, the containment failure locations are at the large pipe penetrations. At pressures above 123 psig, the failure location is at the mid-height region of the containment. The best-estimate containment failure pressure occurs at 127.6 psig due to membrane stresses at the mid-height region of the containment. For the MAAP analyses, the lower bound values of 99 psig are used. The effect of elevated temperature upon the containment capacity was not evaluated, however, the use of the lower bound containment capacity implicitly takes into account the reduction of containment capacity with increased temperature.

The effect of elevated temperatures upon containment electrical and mechanical penetrations was analyzed as a part of the IPE, and a summary of the analyses was provided in response to the NRC RAIs [7]. Mechanical (piping) penetrations do not contain non-metallic seals or gaskets that can be susceptible to potentially high gas temperatures. Personnel locks, equipment hatches and purge lines have non-metallic gaskets. The peak temperature in the region of these penetrations was found to be approximately 450°F for station blackout sequences (without RWST injection), 230°F for small LOCAs, and 200°F for large LOCAs. However, a review of the performance of the non-metallic penetrations and sealant materials indicated that the gasket material can withstand temperatures up to 550°F without leakage.

The Electrical Penetration Assemblies (EPAs) in the WCGS containment were manufactured either by Conax or by Bunker-Ramo. The Conax EPAs use polysulfone and Viton as sealant materials, while Bunker-Ramo use ethylene-propylene rubber. The Conax EPAs have been shown to maintain their integrity at temperatures exceeding 500°F, failure of these EPAs is not a concern.

The Bunker-Ramo EPAs have been tested in the laboratory for up to 347°F. The experimental data (lifetime vs. temperature) was fitted using an Arrhenius plot, and was extended to higher temperatures. It was determined that the EPAs will leak after exposure for about 40 hours to elevated temperatures. However, the leak rate is less than 1×10^{-6} cc/sec which is a negligible rate. For comparison, the normal containment leakage rate is 100 cc/sec. Hence, in summary, failure of electrical and mechanical penetrations due to elevated temperatures, is very unlikely in the WCGS containment.

2.2.4 Containment Isolation Failure

A detailed analysis of the containment isolation system in the WCGS plant is provided in response to the NRC team RAIs [7]. A fault tree was developed for the containment isolation node of the CSET. The isolation valves were divided into three categories (A, B, and C), as per the Westinghouse classification. Isolation failure of the category A valves were addressed

using a scalar value, as per Westinghouse guidelines. No other fault tree modelling was performed for these valves. The containment isolation fault tree for Category B and C valves was evaluated twice, once for the station blackout sequences, and the other time for all initiators except the station blackout sequences. The failure probability of the containment isolation system (for all initiators except the station blackout system) was calculated to be 1.728×10^{-4}. Failure of Category A penetrations contributed to 69% of this value. The containment isolation system fault tree was analyzed separately for station blackout sequences, with the AC power system assumed to fail. The calculated failure probability was 2.31×10^{-3}. Failure of the reactor coolant drain tank discharge isolation contributed to 87% of this value.

Containment bypass was also analyzed as a part of the IPE. All systems interfacing with the RCS were identified and screened to assess the potential for ISLOCA. The following paths for ISLOCA were judged to be potentially significant:

- RHR suction line and suction isolation valves,
- RHR accumulator injection lines and injection supply isolation valves,
- Safety injection pump discharge to cold leg injection,
- Safety injection pump discharge to hot leg recirculation, and
- RHR train A & B safety injection system hot leg recirculation isolation valves.

The initiating frequency of the V-sequence is stated to be 6.11×10^{-8} per reactor year. Steam Generator Tube Rupture (SGTR) sequences were analyzed in the front-end, and a CDF of 6.26×10^{-7} per reactor year was calculated. However, re-analysis of several front-end SGTR sequences using the MAAP code in the back-end analysis, led to the conclusion that core damage was not possible for these sequences within the 24 hour mission time. After an elimination of these sequences, the contribution of the SGTR sequences was found to be reduced to 1.2×10^{-8} per reactor year. Elimination of the SGTR sequences identified by the front-end analysts is a shortcoming of the submittal, since the risk profile of the nuclear plants is dominated by bypass sequences such as the SGTR sequence.

2.2.5 System/Human Response

Although the WCGS CET includes no explicit modelling of operator actions, the licensee stated in response to NRC RAIs [7] that some of the operator actions directed by EOPs were modelled in the IPE. Examples include: spray recirculation and use of low pressure ECCS after depressurization coincident after vessel failure. Operator actions not modelled include opening of PORVs and restarting of RCPs on high core exit temperature. In addition operator actions that may lead to recovery of systems after possible recovery of AC power (after core damage), were not modelled.

The licensee evaluated the negative consequences of the operator actions (as directed by EOPs) [7]. The only action that the licensee identified as possibly having a negative consequence, was the operator action to restart the RCPs, as directed by the EOPs. Loop seal clearance was found to lead to induced SGTR. It was found that if this operator action was modelled, the frequency of bypass sequences was found to increase from 7.3×10^{-8} per reactor year to 2.34×10^{-7} per reactor year.

2.2.6 Radionuclide Release Categories and Characterization

The results of the Wolf Creek CET analyses lead to fourteen end-states, which are further classified into six end-state bins, and they are:

- Leakage (containment intact),
- Late containment overpressure failure,
- Late containment failure and scrubbed fission product release,
- MCCI-induced basemat melt-through,
- Early containment overpressure failure, and
- Early containment failure and scrubbed fission product release

However, the release category definitions are more detailed, and are defined as the following:

A No containment failure within the 48 hour mission time, but containment failure possible if accident management steps are not taken, noble gases and less than 0.1% of the volatiles released.

B Containment bypassed with noble gases and less than 0.1% of the volatiles released.

C Containment bypassed with noble gases and up to 1% of the volatiles released.

D Containment bypassed with noble gases and up to 10% of the volatiles released.

E Containment isolation failure with noble gases and less than 0.1% of the volatiles released.

F Containment isolation failure with noble gases and up to 1% of the volatiles released.

G Containment isolation failure with noble gases and up to 10% of the volatiles released.

H Early containment failure with noble gases and less than 0.1% of the volatiles released.

I Early containment failure with noble gases and up to 1% of the volatiles released.

J Early containment failure with noble gases and up to 10% of the volatiles released.

K Late containment failure with noble gases and less than 0.1% of the volatiles released.

L Late containment failure with noble gases and up to 1% of the volatiles released.

M Late containment failure with noble gases and up to 10% of the volatiles released.

N late containment failure with noble gases, up to 1% of the volatiles, and up to 0.1% of non-volatiles released.

T Containment bypassed with noble gases and more than 10% of the volatiles released.

U Containment isolation failure with noble gases and more than 10% of the volatiles released.

V Early containment failure with noble gases and more than 10% of the volatiles released.

W Late containment failure with noble gases and more than 10% of the volatiles released.

The source terms for the release categories were obtained from MAAP calculations. Each PDS sequence was analyzed using the CET and a number of CET end-states were generated. After all the PDSs were analyzed, each CET end state was found to be composed of a large number of sequences. The sequence which has the largest contribution to the CET end-state was simulated using the MAAP code. In order to generate the source term magnitude corresponding to other release categories that correspond to the same end-state, more MAAP calculations were performed. The number of source term analyses performed were too numerous to list here, and are provided in Table 4.3-6 of the submittal. Table 2 of this TER provides the frequency of the important release categories in the IPE submittal and their frequencies.

Generic Letter 88-20 states that: "any functional sequence that has a core damage frequency greater than or equal to 10^{-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," should be reported by the IPEs. The IPE submittal states that no sequences meeting the reporting criteria, were identified.

Table 2 Frequencies and Conditional Probabilities of WCGS IPE Release Categories

Release Category	Definition	Frequency (Per Reactor Year)	Conditional Probability
S	Leakage, No Containment Failure	2.15E-5	0.59
A	No containment failure	1.25E-5	0.35
K	Late containment failure with less than 0.1% of the volatiles released	1.4E-6	0.038
T	Containment bypassed with greater than 10% of volatiles released	7.3E-8	2E-3
G	Containment isolation failure with greater than 10% of the volatiles released	5E-8	1.4E-3
J	Early containment failure with less than 10% of the volatiles released	3.8E-8	1.1E-3
D	Containment bypassed with less than 10% of the volatiles released	1.2E-8	3.3E-4

2.3 Quantitative Assessment of Accident Progression and Containment Behavior

2.3.1 Severe Accident Progression

The MAAP 3.0B code (version 17.02) was used to evaluate the integrated containment response and the severe accident source terms. The MAAP analyses performed for severe accident analyses and source term quantification are quite detailed, and a total of 12 sequences were analyzed. The choice of sequences to be analyzed was based on the CET analyses, in order to enable the proper quantification of source terms for all important end-states. The choice of analyzed sequences is discussed in Section 4.3 of the submittal, and a listing of all the analyzed accident sequences is provided in Table 4.3-2 (page 4-55) of the submittal. The results of the analyses, including the timing of key events, the maximum amount of hydrogen generated in-vessel, containment pressures and temperatures, and the fission product distribution at the end of calculations, are all provided in Table 4.3.2 for all the accident sequences.

A number of sensitivity studies were performed to study the following phenomena:

1. Hydrogen burn completeness.
2. In-vessel hydrogen generation.
3. Hot leg creep rupture in a high pressure sequence.
4. RPV failure mode.
5. Containment failure pressure and failure area.
6. Volatile fission product release/retention in the primary system.
7. Ex-vessel debris coolability.

A summary of sensitivity analyses performed, the parameters that are varies, and a brief summary of results from these analyses, are all provided in Table 4.4-1 of the submittal.

2.3.2 Dominant Contributors to Containment Failure

The containment failure modes and timings for various accident sequences are provided in Section 4.6.3 and summarized in Section 4.8 of the submittal. Table 3 of this review shows a comparison of the conditional probabilities of the various containment failure modes of the WCGS IPE submittal with the Surry and Zion NUREG-1150 results. All comparisons are made for internal initiating events only.

Table 3 Containment Failure as a Percentage of Total CDF: Comparison With Other PRA Studies

Containment Failure Mode	WCGS IPE	Surry NUREG-1150	Zion NUREG-1150
Early Failure	0.11	0.7	0.5
Late Failure	3.8	5.9	24.0
Bypass (V)	0.17	7.6	0.2
Bypass (SGTR)	0.03	4.6	0.3
Isolation Failure	0.14	NA+	1.0
Intact	95.8	81.2	73.0
Core Damage Frequency, yr^{-1}	$4.2x10^{-5}$*	$4.1x10^{-5}$	$6.2x10^{-5}$

* Includes Flooding

\+ Included as a Part of Early Containment Failure

The WCGS core damage frequency for internal events is comparable to that calculated by NUREG-1150 for Surry and Zion [4,5]. The conditional probability of early containment failure (due to overpressurization) in the WCGS plant is 0.001 and is considerably less than that calculated by the NUREG-1150 analyses for the Zion and Surry plants. This is primarily due to the IPE treatment of the phenomena that threaten the containment integrity at vessel breach, such as DCH, steam explosions, etc. All phenomena that have been shown to lead to early containment failure in other IPE submittals, have been considered, but ruled out as contributors to early containment failure. Late failure as defined in the submittal occurs within 48 hours following accident initiation, and is caused by containment overpressure (either due to steam generation or the accumulation of noncondensibles) or basemat melt-through. The conditional probability of 0.038 calculated for late failure in the IPE submittal is less than the corresponding value calculated by the NUREG-1150 analyses for the Surry plant. The small probability of late containment failure is attributed to the following three reasons. First, the containment has a large cavity floor area, and therefore the submittal assumes that there is a high conditional probability of coolability of debris on the cavity floor by an overlying pool of water. Secondly, the concrete type in the WCGS plant is a basaltic aggregate. The generation of non-condensible gases were found to be very small for this type of concrete, and hence the conditional probability of late overpressure failure is calculated to be low. The third probable cause for this difference is the lack of treatment of AC power recovery in the IPE submittal, which was shown to lead to containment failure in the NUREG-1150 analyses for the Surry plant due to condensation of steam and subsequent hydrogen combustion. The calculated release frequency for the ISLOCA and SGTR sequences in the IPE submittal is also smaller than the NUREG-1150 analyses.

2.3.3 Characterization of Containment Performance

The WCGS IPE considered all the possible contributors to early containment failure, and argued that these processes (DCH, steam explosions, hydrogen combustion, melt attack of liner penetrations, and vessel rocketing) do not lead to early containment failure. The calculated conditional probability of early containment failure is exceedingly small ($\sim$0.001). Late containment failure is primarily driven by overpressurization due to steam generation, and to a lesser extent, due to the buildup of noncondensible gases. Station blackout PDSs are the leading contributors (88%) to late containment failure. However, steam generator tube rupture sequences and ISLOCA sequences are the principal contributors (86.5%) to the releases.

2.3.4 Impact on Equipment Behavior

The impact of the accident progression on equipment performance after core damage was not considered as a part of the CET analyses (or deterministic analyses) in the IPE submittal. However, the licensee did consider the impact of severe accident conditions upon the performance of fan coolers, which are the only equipment located inside the containment, and whose performance is necessary for containment integrity. The licensee stated that the location of the fan coolers precludes the impingement of core debris, since the debris ejected at vessel breach will have to make several 90° turns to be able to reach the fan coolers. Several floor levels and obstructions are present between the cavity and the location of the fan coolers. The fan coolers have been qualified to design basis accident conditions of 384°F and 47 psig. In non-station blackout accident sequences, the long term pressure and temperature remains below these limits. For station blackout sequences, no credit is taken for AC power recovery. The fan coolers are qualified for 100% humidity. Aerosol plugging of the fan cooler units was considered in two phases, one prior to significant core-concrete interactions, and the other during the MCCI phase. The masses of radionuclide species in the containment atmosphere were obtained from the MAAP simulations of the accident sequences. In the early phase of a severe accident (prior to the period of significant MCCI), the only radionuclides present in the containment atmosphere are the volatile species (CsI and CsOH), whose mass is not sufficient to clog the surface area of one RFC (242 m^2). However, the mass of the aerosols generated during MCCI can plug the RFC. As an example, for an accident sequence involving a dry cavity, core-concrete interactions taking place over a period of 48 hours can generated about 5000 kg of aerosols. This correspond to a volume of about 1 m^3. Even if the aerosols generated for this sequence may not completely plug the fan coolers, they can lead to a reduction in air flow. However, it may take several days to completely plug the fan coolers, and hence the licensee believes that accident management strategies should take into account the possibility of aerosol plugging of fan coolers in the late phase of the accident.

2.4 Reducing the Probability of Core Damage and Fission Product Releases

2.4.1 Definition of Vulnerability

The submittal does not define "vulnerability", particularly as related to containment analyses. However, the licensee concluded [7] that there are no back-end vulnerabilities at WCGS. This conclusion is based on the following results of the IPE:

- The peak containment pressure due to DCH and hydrogen combustion is well below the tail-end of the containment fragility curve. In addition, the licensee argues that the phenomena such as steam explosions, direct attack of containment penetrations, and vessel rocketing, cannot pose a threat to the containment integrity.

- A containment walkdown was performed to identify any potential locations of hydrogen accumulation, and no such locations could be found. The licensee concluded that hydrogen deflagration and detonation are not possible containment failure mechanisms at the WCGS.

- A number of accident sequences lead to significant ablation of concrete basemat, and a small conditional probability of containment failure due to overpressurization. However, the conditional probability (~0.04) is so low that MCCI cannot be considered as a vulnerability.

- The bypass sequences are small contributors to the CDF (~0.2%).

In summary, the licensee concluded that there are no vulnerabilities related to the WCGS containment performance.

2.4.2 Plant Modifications

WCNOC has identified modifications or improvements in seven areas, namely, high temperature qualified RCP seal rings, replacement of positive displacement charging pump by a centrifugal charging pump, provision of a switch to restore main feedwater (if auxiliary feedwater fails), study of equipment dependence on room cooling, replacement of emergency procedures associated with loss of component cooling water or service water, identification of procedural or hardware modifications to reduce the CDF due to internal flooding, and implementation of accident management guidelines. The only "improvement" relevant to the containment analyses are the planned implementation of accident management guidelines. Westinghouse Owners Group issued the generic guidelines in 1994. The licensee is committed to an assessment of the SAM capabilities and to implement any enhancements by September, 1997 [7].

2.5 Responses to CPI Program Recommendations

One of the recommendations of the CPI program pertaining [2,6] to PWRs with large dry containments was that the utility should evaluate the IPE results for containment and equipment vulnerabilities to hydrogen combustion (local and global), and point out any need for procedural and/or hardware improvements. The submittal documentation does not explicitly discuss the recommendations of the CPI program. However, in response to the NRC questions, the licensee discussed the recommendations of the CPI program, and their treatment of the CPI program recommendations, which was summarized by a position evaluation summary report [7].

The licensee performed a walkdown of the containment, and could not identify any likely locations for hydrogen pocketing. The licensee identified several openings between the lower compartment, steam generator compartment, and the pressurizer cubicle, to the upper compartment. Hydrogen deflagration in the WCGS containment was studied, and was ruled out and has been ruled out as a contributor to early containment failure. A simplified analysis was performed to evaluate the potential for Deflagration-to-Detonation Transition (DDT). The upper containment was modelled as an unconfined geometry. The lower and annular compartments were modelled as channels. The analysis concluded that flame acceleration and transition to detonation was unlikely in all three compartments. The impact of hydrogen combustion upon the functioning of fan coolers was analyzed, but ruled out on the basis that the resulting temperature transient in the upper compartment can last only a few minutes. Assuming natural convection heat transfer to the fan coolers, the temperature transient from a burn has to last more than ten minutes in order to heatup the fan coolers to the point of damage.

3. CONTRACTOR OBSERVATIONS AND CONCLUSIONS

The back-end portion of the Wolf Creek IPE submittal provides a reasonable amount of information in regard to the subject areas identified in Generic Letter 88-20 and NUREG-1335. This submittal uses the results from the MAAP simulations and open literature to exclude most of the early challenges to the containment integrity. The weakness of the submittal is that the licensee has provided very limited information to exclude most challenges to containment integrity and has excluded several steam generator tube rupture sequences in the back-end analyses.

The important points of the technical evaluation of the WCGS IPE back-end analysis are summarized below:

- The Back-End portion of the IPE supplies a reasonable amount of information regarding the subject areas identified in Generic Letter 88-20.

- The WCGS IPE provides an evaluation of all phenomena of importance to severe accident progression in accordance with Appendix I of the Generic Letter. However, all phenomena that can lead to early containment failure are ruled out as not being a threat to WCGS containment integrity.

- Direct sequence-by-sequence linking of front-end accident sequences to the CET analyses was the methodolcgy used for the back-end analyses. However, only sixty front-end accident sequences were considered for back-end analyses, and twelve MAAP simulations were performed for deterministic analyses. The extent of analyses is rather narrow.

- The depth of treatment of phenomenological uncertainty, and severe accident progression issues, is fairly sparse. The CET consists of only seven nodes, and these nodes are not developed any further. In addition, quantification of these nodes, consists mostly of assigning values of zero or one, based on the results of the front-end analyses.

- Several issues, such as AC power recovery, depressurization after core damage, in-vessel core coolability, operator actions after core damage. recovery of containment cooling systems, etc., have not been considered in the containment analyses. Some of these issues have been considered using sensitivity analyses.

- Steam generator tube rupture sequences have been identified by other IPEs to be the dominant contributor to releases in PWRs with a large dry containment. In the Wolf Creek IPE front-end analyses, SGTR sequences were determined to contribute to 1.5 % of the CDF. The licensee re-analyzed the SGTR sequences using the MAAP code, and determined that four of the five analyzed SGTR sequences did not lead to core damage within the 24 hour mission time. Hence, a majority of the SGTR sequences were excluded from the back-end analyses. Exclusion of the risk-dominant SGTR sequences leads to a reduction in radionuclide release frequency and the magnitude of the release.

- The licensee has addressed the recommendations of the CPI program (GL 88-20, Supplements 1 and 2), as a response to NRC review team questions.

No vulnerabilities have been identified, and the only plant improvement planned on the back-end analyses involves the planned implementation of accident management guidelines. Westinghouse issued the generic guidelines in 1994. The licensee is committed to an assessment of the SAM capabilities and to implement any enhancements by September 1997 [7].

4. REFERENCES

1. "WCGS Nuclear Station Individual Plant Examination Summary Report," prepared by Wolf Creek Nuclear Operating Company (September 1992).

2. NRC Letter to All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities, "Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR §50.54(f)," Generic Letter 88-20, Supplement No. 1, dated August 29, 1989.

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

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

5. "Evaluation of Severe Accident Risks: Zion Unit 1", U. S. Nuclear Regulatory Commission, NUREG/CR-4551, Vol. 7, Part 1 (June 1990).

6. NRC Letter to All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities, "Completion of Containment Performance Improvement Program and Forwarding of Insights for Use in the Individual Plant Examination for Severe Accident Vulnerabilities - Generic Letter No. 88-20 Supplement No. 3 - 10 CFR §50.54(f)," Generic Letter 88-20, Supplement No. 3, dated July 6, 1990.

7. "Response to Request for Additional Information," Enclosure to Letter from Niel S. Carns, Wolf Creek Nuclear Operating Company, to the U.S. NRC (August 1995).

8. M. M. Pilch, et al., "The Probability of Containment Failure by Direct Containment Heating in Zion," NUREG/CR-6075, Supplement 1, Sandia National Laboratories, Albuquerque, NM (1994).

9. M. M. Pilch, et al., "The Probability of Containment Failure by Direct Containment Heating in Zion," NUREG/CR-6109, Sandia National Labaratories, Albuqurque, NM (1994).

10. M. M. Pilch, M. D. Allen, E. W. Klamerus "Resolution of Direct Containment Heating Issue for All Westinghouse Plants With Large Dry Containments or Subatmospheric Containments," Draft, NUREG/CR-6338, Sandia National Labaratories, Albuqurque, NM (June 1995).

APPENDIX A

IPE EVALUATION AND DATA SUMMARY SHEET

PWR Back-End Facts

Plant Name

Wolf Creek Generating Station

Containment Type

Large dry containment.

Unique Containment Features

Large cavity floor area.

Containment design does not facilitate flooding of the reactor cavity.

Unique Vessel Features

None found.

Number of Plant Damage States

58 PDS sequences.

Containment Failure Pressure

127.6 psig (median).

Additional Radionuclide Transport and Retention Structures

Auxiliary building structures have not been credited.

Conditional Probability That The Containment Is Not Isolated

0.0023.

Important Insights

Cavity and instrument tunnel design does not permit debris transport to the upper compartment, and large cavity floor area aids debris coolability.

Unique Safety Features

None identified.

Implemented Plant improvements

No plant improvements were found necessary.

C-Matrix

Direct linking of front-end accident sequences with the CET in the IPE makes it difficult to generate a C-matrix.

APPENDIX C

WOLF CREEK GENERATING STATION INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(HUMAN RELIABILITY ANALYSIS)